Boiling Water Reactor
GE BWR/4
Technology Advanced Manual

Chapter 3.0

Technical Specifications

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List of Attachments

3.0 TECHNICAL SPECIFICATION ORGANIZATION

Learning Objectives :

1. State the purpose of Technical Specifications
2. State the purpose of Specification 3.0.3.

3.0.1 Introduction

The purpose of Technical Specification is to protect the health and safety of the public by imposing limits, operating conditions, and other similar requirements on the facility.

The legal requirements for plant technical specifications are found in 10 CFR 50.36 which states "The technical specifications will be derived from the analyses and evaluation included in the safety analysis report....". Paraphrasing this statement, technical specifications define the limits of plant operation to ensure that the plant is operated within those boundaries established by the Safety Analysis. For example, if the safety analysis uses a maximum reactor coolant system pressure of 1325 psig then a technical specification limit of 1325 psig will be imposed. After the plant's technical specifications have been approved by the Commission, they become part of the licensing document.

3.0.2 Derivation

The format for technical specifications evolves from 10 CFR 50.36 which lists the following categories to be included in technical specifications:

- Safety limits and limiting safety system settings,
- Limiting conditions for operation,
- Surveillance requirements,
- Design features, and
- Administrative controls.

For special items of interest, the NRC issues Regulatory Guides which describe methods acceptable to the NRC staff of implementing specific parts of regulations. One such Regulatory Guide (1.70) provides the STANDARD format and content of Safety Analysis Reports (SARs). This guide specifies seventeen chapters in the SAR, and assigns technical specifications to Chapter 16. Portions of this Regulatory Guide dealing with technical specifications are included below.

3.0.3 Format

There are three technical specification formats that are currently being used. The oldest of these formats is called "custom" technical specifications because the format that was used was decided by the utility. Attachment A illustrates a typical "custom" specification for chemistry. The specification is actually a limiting condition for operation. A limiting condition for operation is defined as a requirement that must be satisfied for the unrestricted operation of the unit. The statements that follow the limiting condition for operation (LCO) are actions that must be taken in the event that the LCO cannot be satisfied. Note that the actions of these statements require a plant shutdown if the LCO cannot reestablished. The bases for the specification follow the limiting condition for operation and its associated action statements. The surveillance, to ensure that the LCO is satisfied, is located in the right hand column across from the LCO.

In the mid seventies, the format for technical specifications was changed to a "standard" format. This format is shown in Attachment B. The standard technical specifications format starts with the LCO statement. Again, the LCO must be satisfied for unrestricted operation. The action statements, i.e., the required actions that must be taken if the condition of the LCO cannot be satisfied, are listed next. The surveillance requirements follow the action statements. Bases for a particular specification are in separate sections of the document.

The third version of technical specifications, NUREG-1433, Revision 1, was issued in April of 1995 and incorporates the cumulative changes resulting from the experience gained from license amendment applications. Many licensees have or plan to convert to these improved Standard Technical Specifications (STS) or to adopt partial improvements to existing technical specifications. NUREG-1433 was the result of extensive public technical meetings and discussions between the Nuclear Regulatory Commission staff and various nuclear power plant licensees, Nuclear Steam Supply System (NSSS) Owners Groups, specifically the GE Owners Group, NSSS vendors, and the Nuclear Energy Institute. The improved STS were developed based on the criteria in the Final Commission Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, dated July 22, 1993. Licensees are encouraged to upgrade their technical specifications consistent with those criteria and conforming, to the extent practical and consistent with the licensing basis for the facility, to Revision 1 to the improved STS. The Commission continues to place the highest priority on requests for complete conversions to improved STS. Licensees adopting portions of the improved STS to existing technical specifications should adopt all related requirements, as applicable, to achieve a high degree of standardization and consistency.

The new improved STS consist of three volumes:

- Technical Specifications,
- Bases, and
- Technical Requirements Manual

The technical specifications volume, illustrated in Attachment C, begins with the LCO followed by the applicability, action, and surveillance sections. The actions sections are divided into three columns (condition, required action, and completion time) while the surveillance sections are divided into two sections (surveillance and frequency). This format is provided to better articulate to the operator the conditions that exist and what must be performed for that condition.

3.0.4 New Revised Standard Technical Specifications

The description that follows is based on the new revised standard technical specification format (the third format discussed above) and will be used in the Advanced Technology and Simulator Courses.

3.0.4.1 Use and Application

This section of technical specifications manual is comprised of four subsections:

- Definitions
- Logical Connectors
- Completion Times
- Frequency

Subsection 1.1 provides defined terms that appear in capitalized type and are applicable throughout technical specifications and bases.

The Logical Connectors, subsection 1.2, explains the meaning of logical connectors and provides examples to illustrate their usage.

The Completion Time, subsection 1.3, establishes the completion time and provides guidance for its use.

The Frequency, subsection 1.4, defines the proper use and application of frequency requirements.

3.0.4.2 Safety Limits

This section of the plant technical specifications establishes the requirements for the protection of the fission product barriers. These requirements are called safety limits. For BWRs, the safety limits are:

- *Thermal Power*, Low Pressure or Low Flow
- *Thermal Power*, High Pressure and High Flow
- Reactor Coolant Pressure
- Reactor Vessel Water Level

When these limits are satisfied, then the fuel cladding and reactor coolant system pressure boundaries are protected during anticipated operational occurrences.

3.0.4.3 LCOs and Surveillance Requirements

Sections 3.0/4.0 are used to establish the ground rules for the remaining portions of technical specifications. One of the most important specifications in this section is 3.0.3, the "motherhood" statement. This specification provides guidance for plant operation when the LCO <u>and</u> its associated action statements cannot be satisfied. For example, one of the ECCS LCOs requires two trains of low pressure systems to be operable. If one train is out of service, operation may continue for some time period. However, if both trains are out of service the actions of specification 3.0.3 must be taken. In summary, when the plant is less conservative than the least conservative technical specification action statement, go to specification 3.0.3. In addition to providing guidance for plant operation in unusual conditions, sections 3.0 also endorses ASME section XI as the testing document for power plant pumps and valves.

The remaining parts of 3/4 specifications deal with individual systems. The following is a listing of the sections or categories and their associated systems:

- 3.1 Reactivity Control Systems
- 3.2 Power Distribution Limits
- 3.3 Instrumentation
- 3.4 Reactor Coolant System
- 3.5 Emergency Core Cooling System
- 3.6 Containment Systems
- 3.7 Plant Systems
- 3.8 Electrical Power Systems
- 3.9 Refueling Operations
- 3.10 Special Operations

3.0.4.4 Design Features

Section 4.0 describes the important design features of the unit. Items such as the cyclic limits of the reactor coolant system and it associated components are listed here. In addition, the emergency plan exclusion and low population areas are shown in this section.

3.0.4.5 Administrative Controls

Administrative controls delineate the management and staff organization, review and audit groups, record and reporting requirements, and procedures required to assure safe plant operation. The administrative organization is addressed in terms of offsite management and onsite staff requirements including the minimum shift crew composition for all plant conditions. The review of safety related matters is conducted by Plant Review Board and the Safety Review Board. Although these are separate groups, they function together in the review and submittal of reports concerning safety matters.

The General Manager shall provide direct executive oversite over all aspects of the plant. The Assistant General Manager-Plant Operations shall be responsible for overall unit operation. Offsite and onsite organizations, in addition to shift manning, are established per administrative control section 6.2.

3.0.5 Bases

The bases for technical specifications requirements is found in a separate BASES manual

3.0.6 Technical Requirements Manual

The Technical Requirements Manual (TRM) contains specifications and operational conveniences, such as lists, cross references, acceptance criteria, and drawings. TRM specifications are contained in Section 3.0 and include operational requirements, surveillance, and required actions for inoperable equipment. Instructions for the use and application of TRM specifications are included at the beginning of Section 3.0

Operational conveniences provide a ready reference to setpoints, lists, and other helpful tools described in plant procedures and programs.

Other plant documents, such as Fire Hazards Analysis, Appendix B, Core Operating Limits Report (COLR), and Offsite Dose Calculation Manual, are not considered part of the TRM, but are included with the TRM as Appendices, and either contain their own rules of usage or are covered by plant documents.

Core Operating Limits Report

Many of the limits discussed in this section must be revised for every core reload cycle. to make a change, a license amendment is required, which must be reviewed by an onsite safety review board and the NRC. This makes any change to these limits a large administrative burden.

NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits for Technical Specifications," dated October 4, 1988, provided guidance for relocating certain cycle dependent core operating limits from Technical Specifications to a Core Operating Limits Report (COLR). The COLR will still be reviewed, but not as a license amendment. Typical core operating limits include the following:

- Control Rod Program Controls
- Average Planar Linear Heat Generation Rate
- Minimum Critical Power Ratio
- Linear Heat Generation Rate

In addition, an entry is added to the definitions to define COLR, and the administrative technical specifications are modified to show the COLR as part of the reporting requirements.

3.0.7 Probability Risk Assessment

Probability Risk Assessment (PRA) of a nuclear power plant provides a tool to quantitatively evaluate the risk implications of Technical Specification (TS) requirements and the risk impact of changes in these requirements. Use of a PRA to evaluate or assess TS requirements and study their modifications is called PRA Informed TS evaluation. Such evaluations are used along with a broad spectrum of considerations which include deterministic analyses, knowledge of lessons learned from operating experiences, and engineering judgments to define or alter TS requirements. When a modification to TS is analyzed using PRA and submitted to the regulatory authority for approval, it is commonly referred to as a PRA-Based or Risk Informed TS submittal. The review and acceptance of the requested modification in the submittal by the regulatory authority constitutes a change in the plant TS.

Assessing the risk impact of a TS change is a useful input in analyzing, reviewing, and accepting the change. Risk-Informed TS submittal evaluations have primarily focused on limiting conditions for operations LCOs) and surveillance requirements. Specifically, PRA-Informed evaluations can be used to address:

- LCO - Identify or define the condition for which a requirement should be defined.
- LCO - Rethink allowed outage time.
- LCO - Determine the required action, i.e., the need for shutdown, additional testing or operability requirements.

PRA-Informed TS submittals primarily deal with permanent changes to TS requirements. The majority of the submittal are motivated to avoid a mode change (plant shutdown).

3.0.8 Exercise

According to technical specifications, when is a recirculation loop considered in operation?

Attachment A

Limiting Conditions for Operation Surveillance Requirements

3.6 Primary System Boundary

B. Coolant chemistry.

1. Prior to startup and at steaming rates less than 100,000 lb/hr, the following limits shall apply.

a. Conductivity, 2.0 μmoho/cm @ 25°F.

b. Chloride, 0.1ppm

2. At steaming rates greater than 100,000 lb/hr, the following limits apply.

a. Conductivity, 2.0 μmho/cm @ 25°F.

b. Chloride, 0.2 ppm

4.6 Primary System Boundary

1. A sample of reactor coolant shall be analyzed:

a. At least every 96 hours for conductivity and chloride ion content.

b. At least every 24 hours during startups, until the steaming rate is greater than 100,000 lb/hr, for conductivity and chloride ion content.

c. At least every 8 hours for conductivity and chloride content when the continuous conductivity monitor is inoperable.

2. During startup prior to pressurizing the reactor above atmospheric pressure, measurements of reactor water quality shall be performed to show conformance with 3.6.B.1 of limiting conditions.

Attachment B

Reactor Coolant System

3/4.4.4 Chemistry

Limiting Condition for Operation

3.4.4 The chemistry of the reactor coolant system shall be maintained within the limits specified in Table 3.4.4-1

Applicability: At all times.

Action:

a. In OPERATIONAL CONDITION 1

1. With the conductivity, chloride concentration or pH exceeding the limit specified in Table 3.4.4-1 for less than 72 hours during one continuous time interval and, for conductivity and chloride concentration, for less than 336 hours per year, but with the conductivity less than 10 µmho/cm at 25 °C and with the chloride concentration less than 0.5 ppm, this need not be reported to the Commission and the provisions of Specification 3.0.4 are not applicable.

2. With the conductivity, chloride concentration or pH exceeding the limit specified in Table 3.4.4-1 for more than 72 hours during one continuous time interval or with the conductivity and chloride concentration exceeding the limit specified in Table 3.4.4-1 for more than 336 hours per year, be in at least STARTUP within the next 6 hours.

3. With the conductivity exceeding 10 µmho/cm at 25 °C or chloride concentration exceeding 0.5 ppm, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 2 and 3, with the conductivity, chloride concentration or pH exceeding the limit specified in Table 3.4.4-1 for more than 48 hours during one continuous time interval, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. At all other times:

1. With the:

a) Conductivity or pH exceeding the limit specified in Table 3.4.4-1, restore the conductivity and pH to within the limit within 72 hours, or
b) Chloride concentration exceeding the limit specified in Table 3.4.4-1, restore the chloride concentration to within the limit within 24 hours, or

c) Perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system. Determine that the structural integrity of the reactor coolant system remains acceptable for continued operation prior to proceeding to OPERATIONAL CONDITION 3.

2. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4 The reactor coolant shall be determined to be within the specified chemistry limit by:

a. Measurement prior to pressurizing the reactor during each startup, if not performed within the previous 72 hours.

b. Analyzing a sample of the reactor coolant for:

1. chlorides at least once per:

a) 72 hours, and

b) 8 hours whenever conductivity is greater than the limit in Table 3.4.4-1.

c. Continuously recording the conductivity of the reactor coolant, or, when the continuous recording conductivity monitor is inoperable, obtaining an in-line conductivity measurement at least once per:

1. 4 hours in OPERATIONAL CONDITIONS 1, and 3, and

2. 24 hours at all other times.

d. Performance of a CHANNEL CHECK of the continuous conductivity monitor with an in-line flow cell at least once per:

1. 7 days, and

2. 24 hours whenever conductivity is greater than the limit in Table 3.4.4-1.

Attachment C

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIS) SYSTEM

3.5.1 ECCS - Operating

LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of seven safety/relief valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3, except high pressure coolant injection (HPCI) and ADS valves are not required to be OPERABLE with reactor steam dome pressure ≤ 150 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One low pressure ECCS injection/spray subsystem inoperable.	A.1 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	12 hours
	AND	
	B.2 Be in MODE 4.	36 hours
C. HPCI System inoperable.	C.1 Verify by administrative means RCIC System is OPERABLE.	1 hour
	AND	
	C.2 Restore HPCI System to OPERABLE status.	14 days

(Continued)

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3.1 CONTROL ROD PROBLEMS

Learning Objectives:

1. State the requirements for Technical Specifications and explain the significance of Limiting Condition for Operation as applied to control rod operablilty, control rod scram times, and Rod Worth Minimizer operability.

2. When given an initial set of operating conditions, the student will be able to use the format and content of the Technical Specifications to identify the applicable plant/or operator response.

3.1.1 Introduction

Control Rods are analyzed to bring the reactor subcritical at a rate fast enough to prevent the OLMCPR from becoming less than the fuel cladding integrity MCPR Safety Limit during the limiting power transient analyzed in Accident analysis section of the FSAR. This analysis shows that the negative reactivity rates resulting from the scram, with the average response of all the drives as required by Technical Specifications, provides satisfactory core protection for the most limiting transient. In addition to APLHGR and the 1% plastic strain fuel design limit.

The standby liquid control system (SLC) provides a backup reactivity control capability to the control rods. The original design basis for the SLC system is to provide a soluble boron concentration to the reactor vessel sufficient to bring the reactor to a cold shutdown condition. In addition to the original design basis, the system must also satisfy the requirements of the ATWS Rule 10 CFR 50.62 paragraph (c) (4), which requires that the system have a control capacity equivalent to that for a system with an injection rate of 86 gpm of 13 weight percent unenriched sodium pentaborate, normalized to a 251 inch diameter reactor vessel.

The term "equivalent reactivity control capacity" refers to the rate at which the boron isotope B^{10} is injected into the reactor core. The SLC system meets this requirement by using a sodium pentaborate solution enriched with a higher concentration of B^{10} isotope.

3.1.2 Shutdown Margin

A sufficient shutdown margin ensures that:

1. the reactor can be made subcritical from all operating conditions;
2. the reactivity transients associated with postulated accident conditions are controllable within acceptable limits; and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

3.1.3 Control Rods

The specifications for control rods ensure that:

1. the minimum shutdown margin is maintained;
2. the control rod insertion times are consistent with those used in the accident analysis; and
3. the potential effects of the rod drop accident are limited.

Limitations on inoperable rods is set so that the resultant effect on total rod worth and scram shape will be kept to a minimum. For a control rod to be considered inoperable, one of the following conditions must exist:

- Immovable due to excessive friction or mechanical interference, or known to be untrippable.
- Unable to meet scram times
- Scram accumulators inoperable
- Uncoupled control rod
- RPIS (rod position can not be determined).
- Not in BPWS when required

Requirements for the various scram time measurements ensure that any indication of systematic problems with control rod drives will be investigated on a timely basis.

Control rods with inoperable accumulators are declared slow or inoperable. The specifications prevents a pattern of inoperable accumulators that would result in less reactivity insertion on a scram than has been analyzed even though control rods with inoperable accumulators may still be inserted with normal drive pressure. Operability of the accumulator is based on maintaining adequate accumulator pressure. When one control rod scram accumulator becomes inoperable and the reactor pressure is >900 psig, the control rod may declared " slow", since the control rod will still scram at the reactor operating pressure but may not satisfy the scram times.

Control rod coupling integrity is required to ensure compliance with the analysis on the rod drop accident. Control rod position may be determined by the use of operable indicators, by moving control rods to a position with an operable indicator, or use other appropriate methods.

To ensure that the control rod patterns can be followed and other parameters are within their limits, the control rod position information system must be operable.

3.1.4 Control Rod Program Controls

Control rod withdrawal and insertion sequences are established to assure that the maximum in sequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle will not be worth enough to cause fuel enthalpy to exceed 280 cal/gm for any postulated rod drop accident

Limitations on inoperable rods is set such that the resultant effect on total rod worth and scram shape will be kept to a minimum. No control rod, if dropped when thermal power is greater than 10%, would exceed 280 cal/gm. Therefore, requiring the RWM to be operable below 10% of rated thermal power provides adequate control.

3.1-5 CRD Testing

Diagnostic testing is the selective analytical testing of specific CRD mechanisms and associated HCUs based on a prior analysis of drive performance and base test data. Tests consists of CRD scram time testing, scram valve timing, stall flow, differential pressure tests, normal drive speeds, and electrical tests. Diagnostic testing limits maintenance outage time and problems with LCO requirements in technical specifications.

Testing will:

- Minimize corrective maintenance to drives with pre-analyzed need, thus maximum utilization of maintenance time.
- Minimizes CRD operational problems in future operating cycles, maximizing plant availability and flexibility.

If a CRD fails to respond to the normal insert/withdraw command signals, notch-in or out of "00", or exhibit scram problems, a differential pressure test should be performed. An analysis of traces generated by measuring the dp changes with an oscilloscope can isolate such faults as:

- CRD mechanical malfunction
- Improper operation of HCU directional control valves (leakage, blockage)
- Improper RMCS timer operation
- Unbalanced hydraulic system (stabilizing valves, flow and pressure control)
- Scram valve leakage
- Air in hydraulic lines
- Improper electrical relay operation

When it is initially determined that an analysis is needed, the following steps should be taken:

- Install testing equipment.
- Apply notch-in signal.

Use a camera to photograph the oscilloscope trace for documentation purposes and as a possible trouble shooting aid.

Normal Notch in

Figure 3.1-1 illustrates a notch in of a control rod drive. A surge pressure of approximately 140 psid is applied until the drive begins moving and drops to about 80 psi to maintain movement.

Air in System

Anytime the control rod drive system is open for maintenance a potential for trapping air in the system exists. In addition, accumulation of air from the CRD water supply over a period of time can occur.

Air in the CRD hydraulic system can result in the following problems:

- Loss of response at directional control valve switching points during the notch out cycle when the volume of air in the supply piping to the Po side increases.
- Loss of driving pressure dp response occurs during a notch-in or notch-out cycle when the volume of air in the supply piping to the Pu side increases. With only 35 in^3 of air trapped in the supply piping a failure to notch can occur.
- Air in the CRD hydraulic system piping can cause breakage of internal drive seals and primary stop piston seals.
- Oxygen is also a contributing cause of intergranular cracking.

Figure 3.1-2 illustrates a control rod being notched out from position 24 with air trapped in piping to the Po side. Note the loss of dp response at directional control valve switching points and the accumulator discharge effect occurring during the settle function.

Figure 3.1-3 illustrates a control rod with insufficient differential pressure to insert the drive. Several problems could cause a low differential pressure, however, in most cases the problem is associated with the hydraulic control unit. Some of the most obvious reasons are listed below.

- plugged filters
- failed closed insert directional control valve
- failed open withdraw directional control valve
- HCU valve line-up not correct
- severe seal damage to drive mechanism
- various electrical malfunctions that could prevent proper valve sequencing.

3.1.6 Exercise

You are a resident inspector at a BWR/4 plant that has just completed a 125 day refueling outage. When you arrive at the station, the post outage plant startup is in progress. You proceed to the control room and review the shift supervisor's log. The following entries are recorded.

- Commenced a reactor startup, mode switch placed in startup/hot standby position.
- reactor critical, critical data taken
- at the point of adding heat
- Rod Worth Minimizer failure, Ops supervisor informed.
- Mode switch placed in run position.
- Plant chemist reports SLC concentration at 6.3% with a volume of 3000 gal.
- Paralleled to grid.
- Scram testing commenced.

While reading the log you hear the reactor operator inform the STA that rod 10-43 will not move.

Consult Technical Specifications, to determine control rod operability requirements, scram times requirements, and Rod Worth Minimizer requirements.

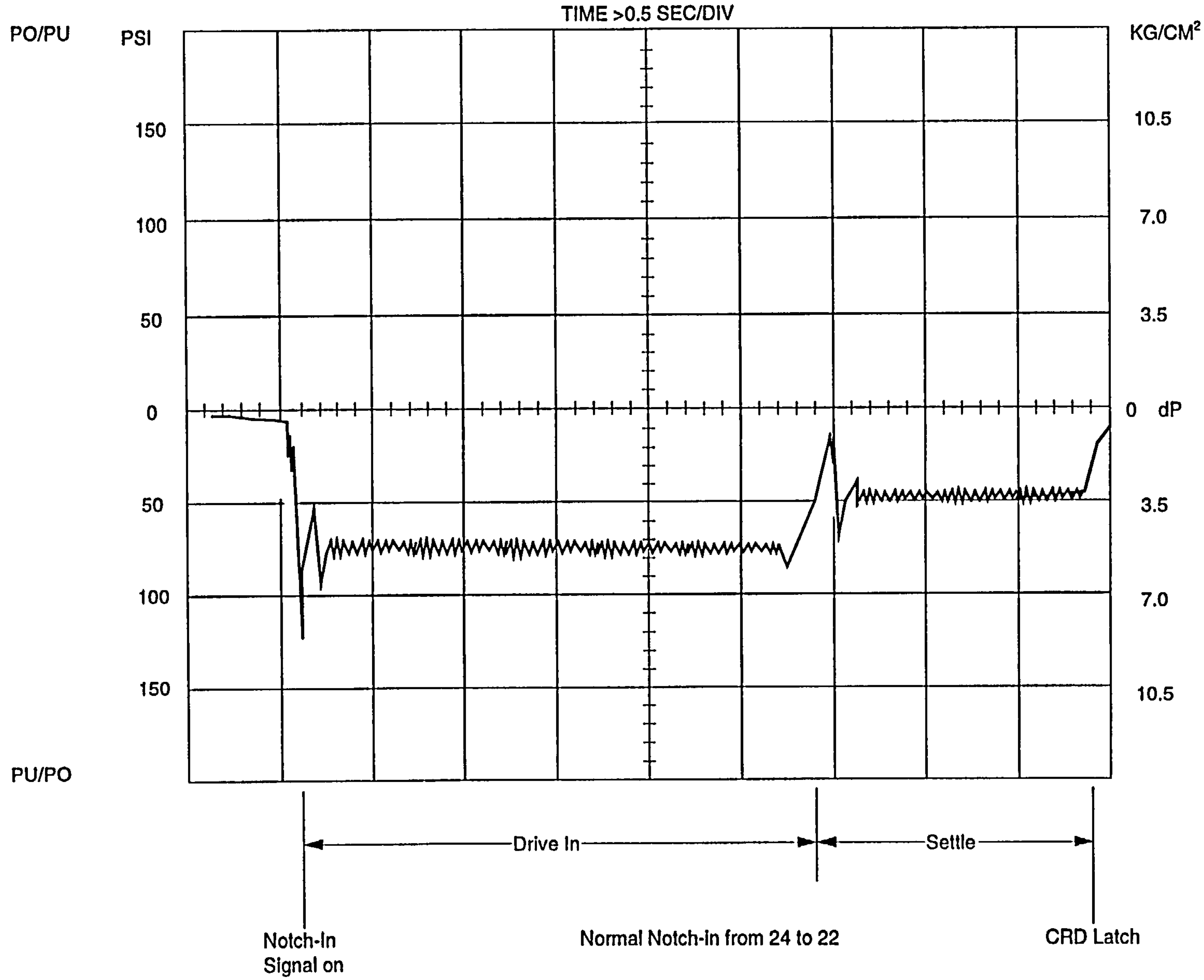

Figure 3.1-1 Normal Notch In

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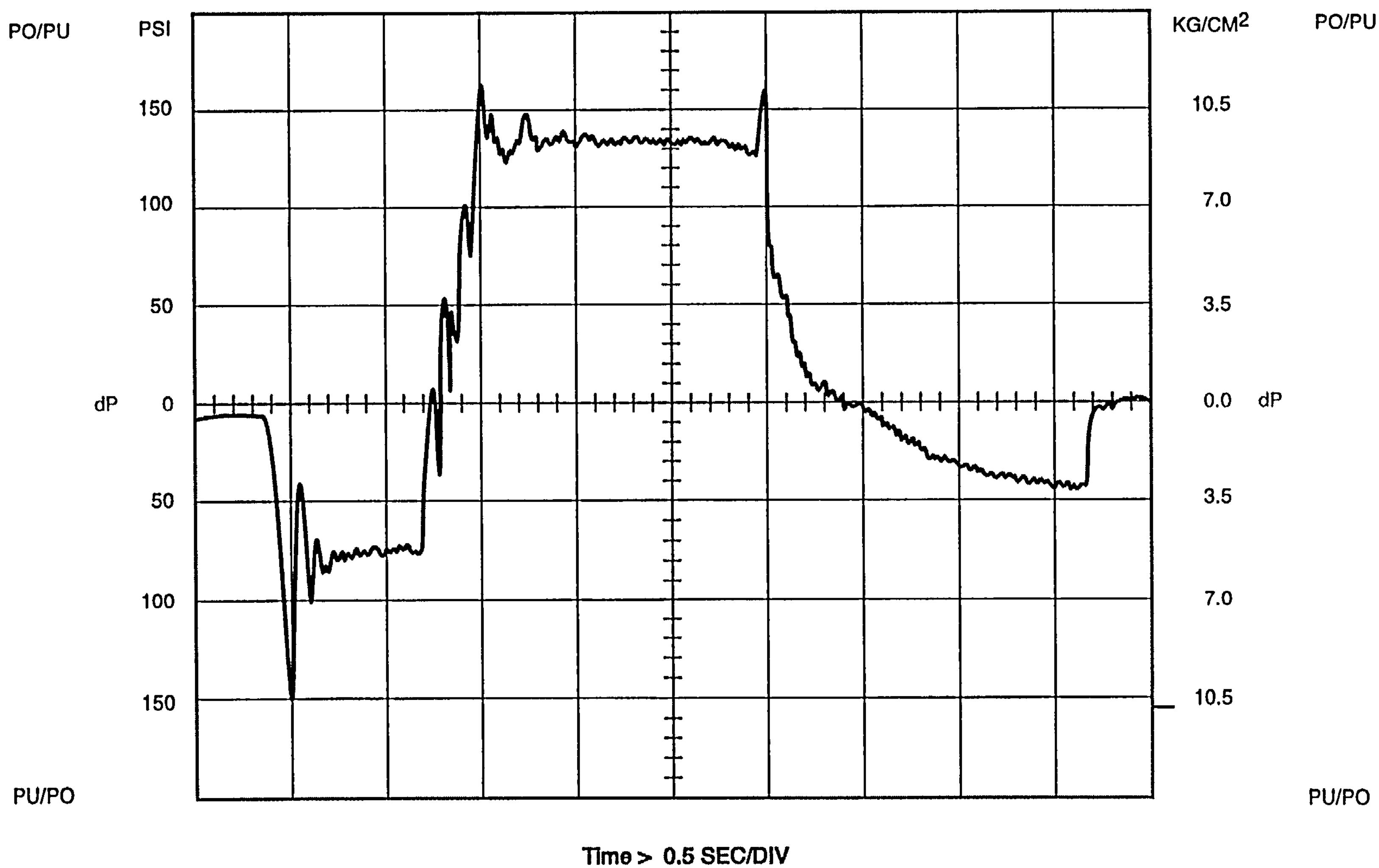
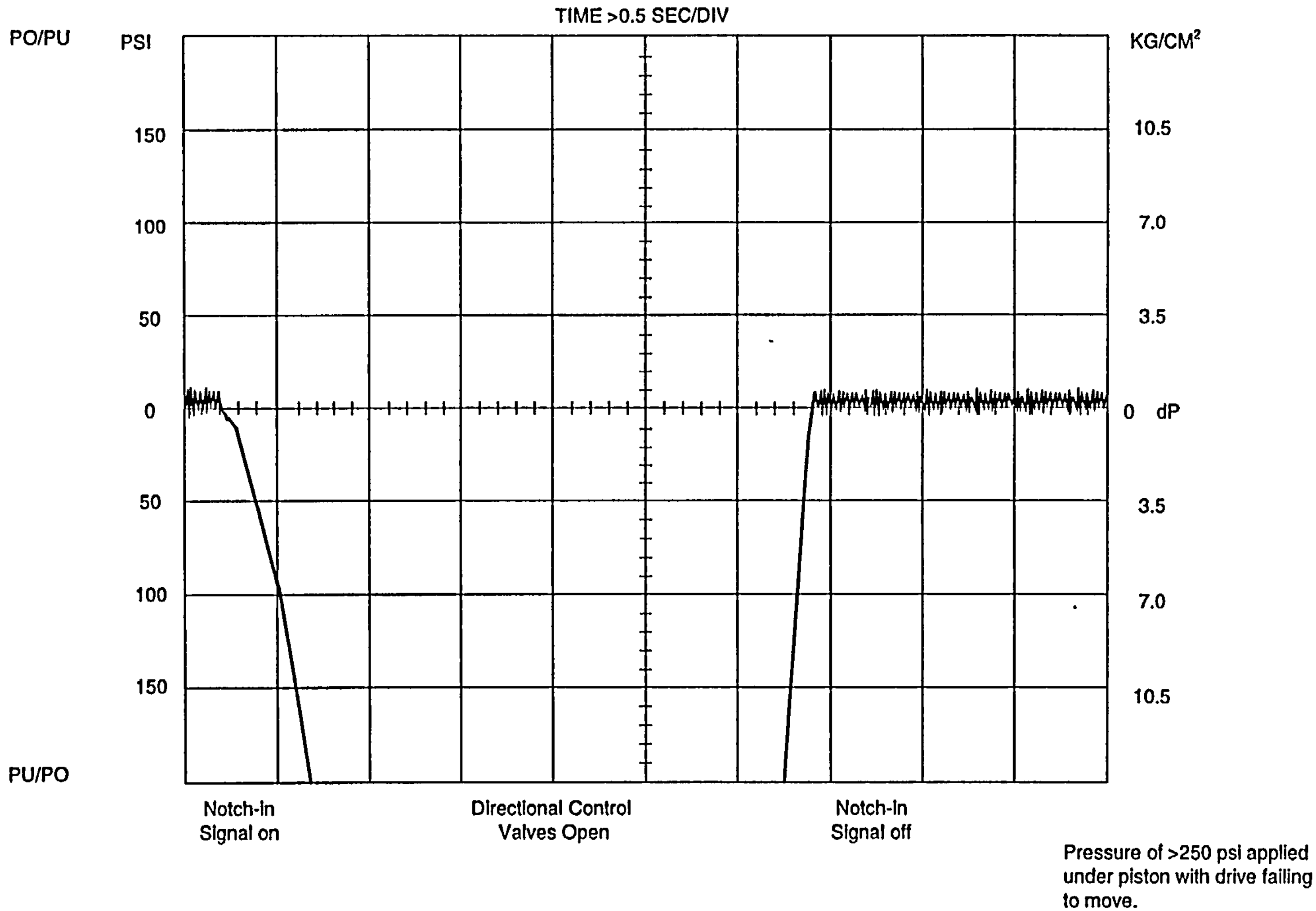


Figure 3.1-2 Notch-Out From Position 24 With Air Trapped In Piping To CRD Po Side



Abnormal notch in from 24 to 22

Figure 3.1-3 High Differental Pressure

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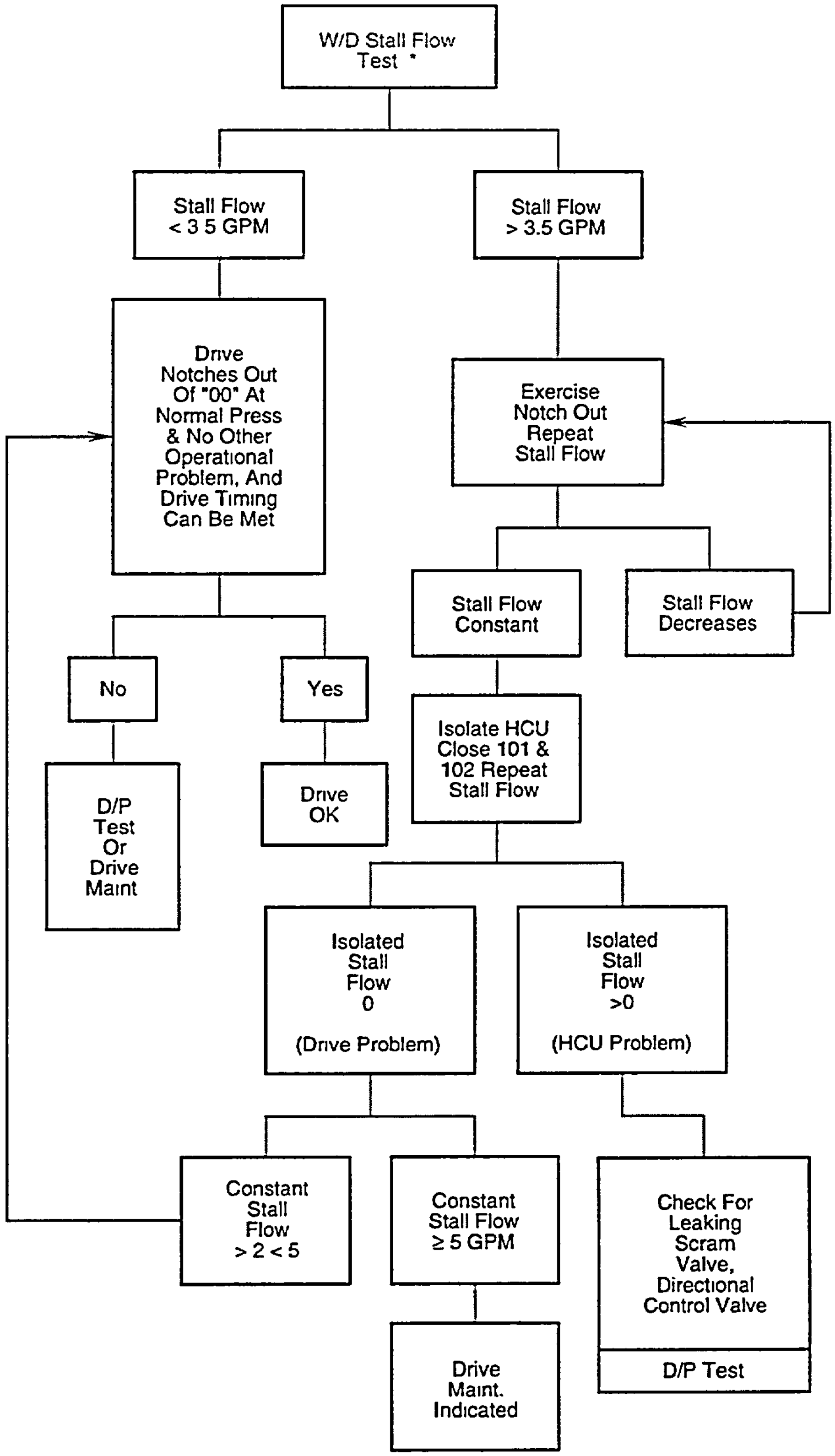
This chart is intended only as a troubleshooting guide and information obtained as a result of its use should not be considered conclusive as to the condition of a drive



* Normal W/D Stall Flow For New Drives Is Between 1 And 2 GPM

Figure 3.1-4 Trouble Shooting Guide

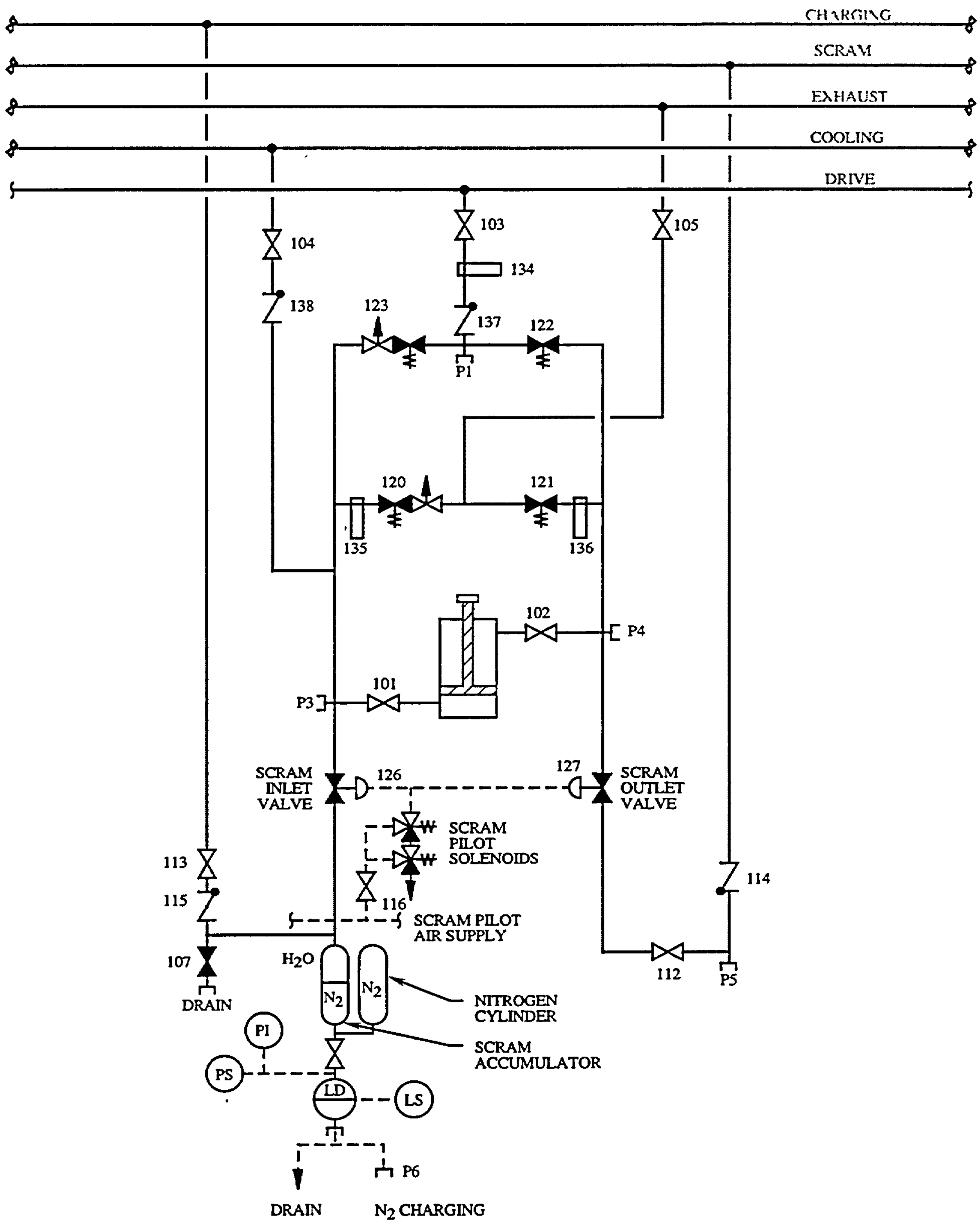


Figure 3.1-5 CRD Hydraulic Control Unit

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3.2 THERMAL LIMITS

Learning Objectives:

1. State the requirements for Technical Specifications and explain the significance of Limiting Condition for Operation as applied to Safety Limits and Power Distribution Limits.

2. When given an initial set of operating conditions, the student will be able to use the format and content of the Technical Specifications to identify the applicable plant/or operator response.

3.2.1 Introduction

Limits on plant operation are established to assure the plant can be safely operated and not pose any undue risk to the health and safety of the public. This is accomplished by demonstrating that radioactive releases from the plants during normal operation, abnormal operation, and postulated accidents meet applicable regulations within conservative limits. These limits are specified by the Technical Specifications to prevent fuel damage from occurring.

The objective for establishing thermal limits for normal operation and transient events is to maintain the integrity of the fuel cladding. This is done by limiting fuel rod power density to avoid over stressing the fuel cladding due to pellet-clad differential expansion, and to avoid centerline melting. Transition boiling must also be prevented to avoid cladding damage due to overheating.

The objective for establishing a thermal limit for postulated accidents is to maintain core geometry by minimizing gross cladding failures. This failure could occur during a Design Basis Accident Loss-of-Coolant Accident (DBA LOCA) where the loss of coolant causes a severe heatup of the cladding. Under worst case conditions, the fuel could suffer gross fragmentation failure due to the quenching action of the Emergency Core Cooling System (ECCS) when the core is reflooded. This is prevented by limiting the stored heat in the fuel, thereby limiting cladding heatup during a LOCA.

The thermal limits established for these purposes are the ECCS/LOCA limit, the thermal-mechanical limit, and the minimum critical power ratio (MCPR) limit (Figure 3.2-1).

3.2.2 Thermal-Mechanical Limit

The thermal expansion rates of the UO_2 pellets and zircalloy cladding are different. The relative expansion arises from several sources:

- UO_2 fuel thermal expansion coefficient is approximately twice that of zircalloy.
- Fuel pellets operate at higher temperature than the cladding.
- Fuel pellets undergo irradiation growth as they are exposed.
- Fuel pellets crack and redistribute toward the cladding when under thermal stress.

This contact places stress on the cladding, If the stress exceeds the yield stress of the cladding material, the cladding will crack. Cladding cracking due to differential expansion of the pellet and clad is prevented by placing a limit on the peak fuel pin power level which would result in 1% plastic strain on the clad. The 1% plastic strain limit itself is conservative. It has been shown that even at the design end of life exposure on the fuel cladding (most brittle condition), greater than 1% plastic strain on the clad is required for cladding failure. This limit is called the *mechanical* limit.

Another limit on peak fuel pin power prevents centerline melting. During transient conditions, fuel pellet overpower occurs which must be limited to prevent centerline melting. This limit is called the fuel pellet *thermal* limit.

These two peak kw/ft limits are usually grouped together and called the *thermal-mechanical* limit.

Technical Specifications do not directly limit peaking factors, therefore it is possible to operate the core at low power with a high TPF. If power is then increased by flow it is possible to exceed the thermal-mechanical (LHGR) limit. To prevent this problem from occurring, Technical Specifications requires the APRM scram and rod block settings to be adjusted whenever the fraction of rated power (FRP) is greater than the core maximum fraction of limiting power density (CMFLPD). Where the fractional limiting power density is the actual LHGR divided by the design value. The process computer automatically calculates the CMFLPD along with the periodic core performance log.

3.2.2.1 Steady State Thermal-Mechanical Limit

The peak kw/ft limit is exposure dependent. The combined steady state limit is determined by the most limiting thermal or mechanical limit and is set by the manufacturer for each fuel type. The limit at zero exposure is 13.4 kw/ft for all GE fuel except GE8B, GE9B, GE10B, and GE11B, which have a 14.4 kw/ft limit. These limits start to decrease after approximately 15,000MWd/st. The zero exposure limit is called Linear Heat Generation Rate (LHGR) limit, by most Technical Specifications.

3.2.3 APLHGR Limit

In the event of a DBA LOCA, the heat stored in the fuel at the time of the event could significantly damage the fuel cladding. The criteria that must be satisfied during this event are given in 10 CFR 50.46. During the DBA LOCA, the core region is voided of liquid in a relatively short time (less than 30 seconds). With no coolant, the only mechanism for heat removal from the cladding is radiative heat loss. The elevated fuel cladding temperatures cause an increase in the rate of oxidation of the zircalloy by the high temperature steam. The chemical reaction becomes self-sustaining at approximately 2800 °F. Formation of zirconium oxide causes the cladding to become brittle. If the cladding temperature increases sufficiently (greater than 2200 °F) for extended length of time, the hot brittle fuel cladding could fragment by the quenching action when the ECCSs reflood the core.

Because the LHGR is used to determine APLHGR the ECCS/LOCA limit and the thermal-mechanical limit can be combined into one number. The result is an exposure dependent curve of Maximum Average Planar Linear Heat Generation Rate limit ($MAPLHGR_{limit}$).

Current GE BWR MAPLHGR limits (as a function of exposure) are based on the most limiting value of either the ECCS/LOCA limits or the thermal-mechanical design limits. Since the thermal-mechanical design limit is included in the determination of the $MAPLHGR_{limit}$, it can not be exceeded if the $MAPLHGR_{limit}$ is met. General Electric has proposed and the NRC has agreed that the separate specification of the steady state thermal-mechanical limit in the Technical Specifications is redundant and can be eliminated. The $MAPLHGR_{limit}$ will continue to provide assurance that the limits in 10 CFR 50.46 will not be exceeded, and that the fuel design analysis limits defined in NEDE 24011-P-A (GESTAR-II) will be met. The steady state thermal-mechanical limits are incorporated by reference into GESTAR-II.

Figure 3.2-2 is an example of a typical $MAPLHGR_{limit}$ curve for 8x8 fuel. The general shape of the curve is produced by using the most limiting kw/ft value calculated for each of the previous criteria. A number of different factors contribute to the change in the curve:

- Changes in local peaking factor with exposure.

- Buildup of fission product gases inside the fuel rod increase the internal gas pressure and decrease the thermal conductivity of the gas pressure.

- Fuel pellet densification

- Response of the plant ECCSs during the DBA LOCA.

The last concern requires plant specific analysis. Therefore, the curves may be slightly different for different plants even though the fuel type is the same.

PCT following a LOCA is primarily a function of the average heat generation rate of all the rods in a fuel assembly at any axial location and is dependent secondarily on the rod-to-rod power distribution within an assembly. The peak cladding temperature is calculated assuming an LHGR for the highest powered rod less than or equal to the design LHGR corrected for fuel densification.

The calculational procedure used to establish the APLHGR limits for Technical Specification is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric calculational models which are consistent with the requirements in Appendix K to 10 CFR 50. The LOCA analysis was performed utilizing the new improved calculational model, SAFER/GESTR-LOCA. The analysis demonstrated that LOCAs do not limit the operation of the fuel. Therefore, the APLHGR limits for the fuel types shown in the Core Operating Limits Report are based on the fuel thermal-mechanical design criteria.

3.2.3.1 Modifications Associated with the APLHGR Limit

A flow dependent correction factor is applied to rated conditions APLHGR to assure that the 2200 °F PCT limit is complied with during a LOCA initiated from less than rated core flow. In addition, other power and flow dependent corrections are applied to rated conditions APLHGR limit to assure that fuel thermal-mechanical design criteria are preserved during abnormal transients initiated from off-rated conditions. The MAPFACs are defined separately as a function of power and flow.

$$MAPLHGR \times MAPFAC_p = MAPLHGR_p$$
$$MAPLHGR \times MAPFAC_f = MAPLHGR_f$$

The MAPLHGR is taken from a figure similar to Figure 3.2-2 for each fuel type, and the MAPFACs are taken from Figures 3.2-3 and 3.2-4. $MAPFAC_p$ is usually determined from feedwater controller failure event results. $MAPFAC_f$ is usually determined by the recirculation pump runout event results. Below P_{bypass}, there is significant sensitivity to core flow during transients. P_{bypass} is defined as the power level which a reactor scram on turbine stop valve position/turbine control valve fast closure is bypassed. For this reason the $MAPFAC_p$ is further defined separately for a high flow (> 50% core flow) and a low flow condition (≤ 50% core flow). Below 25% rated power, surveillance of thermal limits are not required, due to the very large operating margins. Therefore, the $MAPFAC_p$ graph is not addressed below 25% power.

For single loop operation, a multiplication factor is applied to the rated conditions APLHGR power and flow dependent correction factors and the limiting values for APLHGR for each fuel type used in a particular cycle.

After the correction factors have been applied, the lowest MAPLHGR value is the $MAPLHGR_{limit}$ for that power and flow. These calculations are performed by the process computer.

3.2.3.2 MAPLHGR Determination

The process computer calculates the total power produced in every node in the core. A portion of the power produced in a node is produced outside the fuel pins by gamma heating and neutron moderation. This power is divided into two parts: The fraction of the total nodal power that is produced outside the fuel channel in the leakage flow (FLK) and the fraction of the total nodal power produced in the channel that is not conducted through the cladding (FCH). Therefore, for comparison to the $MAPLHGR_{limit}$, the average power density in a node is calculated as follows:

$$MAPLHGR = \frac{Pnode \times (1 - FLK - FCH) \times 1000 \times PTOPF}{NRB \times DZSEG}$$

where:

- FLK = Fraction of core power deposited in leakage flow
- FCH = Fraction of core power deposited in active channel flow by methods other than convection.
- PTOPF = Fraction of core thermal power generated in the bottom 144 inches of fuel.
- NRB = Number of fuel rods per bundle.
- DZSEG = Fuel segment length (ft) = 0.5
- Pnode = Power produced in the node (MW)

3.2.2.3 Peak kw/ft Determination

The process computer calculates the peak kw/ft value for each fuel bundle node in the core. The full core power distribution program (P1) edits these values as MRPD (Maximum Rod Power Density).

MRPD = MAPLHGR x FLOP

FLOP = Maximum rod power/average rod power in across section of fuel segment (local peaking factor).

Once these peak nodal kw/ft values are calculated, the computer compares these to the zero exposure steady state thermal-mechanical limit of 13.4 kw/ft (or 14.4 kw/ft for GE8B, GE9B, GE10B, and GE11B). The process computer calls the steady state thermal-mechanical limit RPDLIM. The ratio of MRPD to RPDLIM is called Fraction of Limiting Power Density (FLPD). As long as the largest value of FLPD is less than one, we are assured that we have not exceeded the thermal limit.

3.2.4 CPR Safety Limit

Critical power is the fuel bundle power required to cause transition boiling somewhere in the bundle. The critical power ratio (CPR) of a fuel bundle is the ratio of its critical power to its actual operating bundle power. The minimum value of CPR for all fuel bundles in the core is the Minimum critical power ratio (MCPR) and represents the bundle which is the closest to transition boiling. MCPR limits are imposed to avoid fuel damage due to severe overheating of the cladding.

The required Operating Limit MCPRs (OLMCPRs) at steady state operating conditions are derived from the established fuel cladding integrity Safety Limit MCPR of 1.06 for two-loop operation and 1.07 for single-loop operation, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting as given in Technical Specifications. The steady state MCPR thermal limit is derived from the single design basis requirement:

Transients caused by single operator error or equipment malfunction shall be limited so that, considering uncertainties in monitoring the core operating state, at least 99.9% of the fuel rods are expected to avoid boiling transition.

3.2.4.1 Modifications Associated with the MCPR Limit

The current licensing basis approved with the GENESIS/ODYN models for calculating the OLMCPR for pressurization events is performed in accordance with either or both of two methods known as Option A and Option B. These currently used options are summarized below:

Option A

This approach is comprised of the two-step calculation which follows:

1. The pressurization transient is analyzed using the GENESIS/ODYN models to obtain the change in the critical power ratio (ΔCPR) for the core. Conservative input parameters are used in the analysis, (e.g. scram speed per Technical Specifications).

2. The licensing basis OLMCPR is given as OLMCPR = 1.044 (Safety Limit CPR + ΔCPR).

Option B

This procedure provides for statistical determination of the pressurization transient ΔCPR/(Safety Limit + ΔCPR) such that there is a 95% probability with 95% confidence (95/95) that the event will not cause the critical power ratio to fall below the MCPR Safety Limit. This approach can be satisfied in one of two ways:

1. A *plant-specific* statistical analysis can be performed per the approved statistical methodology procedures to determine the 95/95 ΔCPR/(Safety Limit + ΔCPR); or
2. Generic ΔCPR/(Safety Limit + ΔCPR) Statistical Adjustment Factors (SAF) for grouping of similar type plants can be applied to plant-specific calculations to derive the 95/95 ΔCPR/(Safety Limit + ΔCPR) value.

Utilities using Option B must demonstrate that their plant's scram speed distribution (τ_{ave}) is consistent with that used in the statistical analysis (τ_B). This is accomplished through an approved Technical Specification which requires testing and allows adjustment of the operating limit MCPR if the scram speed is outside the assumed distribution.

The GEMINI/ODYN set of methods has been compared against actual test data. The results of the comparison indicate an improvement in prediction accuracy with GEMINI/ ODYN models. The true 95/95 ΔCPR/(Safety Limit + ΔCPR) will be determined using the same fundamental approach established for the current GENESIS/ ODYN Option B and accounting for the improvement in prediction accuracy. The resulting procedure, which will be used with the GEMINI /ODYN models, simplifies the current two option approach into one.

Licensing analyses accomplished with GEMINI/ODYN models will permit plants to operate under a single set of MCPR limits if scram speed compliance procedures identical to those in current plant Technical Specifications are followed. If scram speed compliance is not demonstrated, more conservative MCPR operating limits must be met. The statistical determination of the transient ΔCPR/(Safety Limit + ΔCPR) factor for the pressurization event will continue to assure 95% probability with 95% confidence that the critical power will not fall below the MCPR Safety Limit.

The Technical Specification limit will be determined from the following general equation:

$$OLMCPR_{Tech.Spec} = OLMCPR_{95/95} + \frac{\tau_{avg} - \tau_s}{\tau_A - \tau_B}(\Delta OLMCPR)$$

where:

ΔOLMCPR = factors derived by the new methodology.

$OLMCPR_{95/95} = \Delta CPR_{95/95}$ + MCPR Safety Limit

For plants that demonstrate scram speed compliance (i.e. $\tau_{ave} \leq \tau_B$) using the NRC-approved procedures, the specification limit becomes:

$OLMCPR_{TechSpec} = OLMCPR_{95/95}$ (for $\tau_{ave} \leq \tau_B$)

If scram speed compliance is not demonstrated by a plant or if a plant chooses not to perform the scram speed compliance procedures (i.e. $\tau_{ave} \leq \tau_B$), then a more conservative limit must be used.

The actual operating limit will be a straight-line interpolation between these two values dependent on the results of scram speed testing, Figure 3.2-5.

At less than rated power conditions, transients such as rod withdrawal errors, feedwater controller failures, or recirculation pump runouts become limiting. For this reason, the OLMCPR is raised to compensate for such transients. These operating limits are:

$MCPR_f$ = a flow biased MCPR operating limit

$MCPR_p$ = a power biased MCPR operating limit (K_p power adjustment factor)

A flow adjusted factor (K_f) increases the CPR operating limit at core flows less than rated (Figure 3.2-6). The upper curve is used when operating in the automatic flow control mode to prevent violation of the OLMCPR if flow increases to the maximum flow rate allowed by the recirculation system. The lower curves are used when operating in the manual flow control mode to prevent violation of the safety limit MCPR if flow increases to the maximum flow rate allowed by the recirculation system.

When operating below P_{bypass} the severity of a limiting event becomes significantly sensitive to the initial flow at which the transient begins. A high initial flow is more limiting. Therefore, to prevent application of the more conservative high flow limits to a typical low flow startup condition, the $MCPR_p$ is further defined for high flow (> 50% core flow) and low flow conditions (≤ 50% core flow). The 50% cutoff for flow is a conservative value.

Since the initial core flow below P_{Bypass} affects the severity of the transient, the value taken from Figure 3.2-7 is the $MCPR_p$ and not the correction factor K_p. Below P_{Bypass}, the severity of events such as Load Reject without bypass of Turbine Trip without bypass can exceed that of a feedwater controller failure.

When operating at rated power and flow conditions, the OLMCPR is the limiting value for MCPR. However, at less than rated power and flow conditions, the $MCPR_f$ and $MCPR_p$ are determined and the largest value of the two becomes the OLMCPR for that power and flow condition.

The process computer calculates $MCPR_{limit}$ where:

$$MCPR_{limit} = \max[(K_p \times OLMCPR), MCPR_f].$$

$$OLMCPR = 1.32$$

3.2.5 Exercise

A concerned nuclear engineer trainee at a facility expresses his concerns that the facility may not be operating within the thermal limits as defined by Technical Specifications.

The data available to you consists of the following:

- 2500 MW_{th} core operating at 90% CTP
- A GE9B (62 fuel rods, 150in. active fuel length) bundle producing 4.5 MW_{th}
- Node 16 producing 0.38MW_{th} (Uncontrolled)
- Critical power for the GE9B bundle is 6.5 MW_{th}
- Core Flow = 80%; NBUN = 560 bundles

- PTOPF = 0.995
- FLK + FCH = 0.04
- Attachment 1 (Last Core Operating Limits Report for the plant)
- Bundle exposure of 25 GWd/st
- Operating limit MCPR = 1.32

Assuming that all of the information given to you is accurate, make the appropriate calculations and determine if the thermal limits have been exceeded.

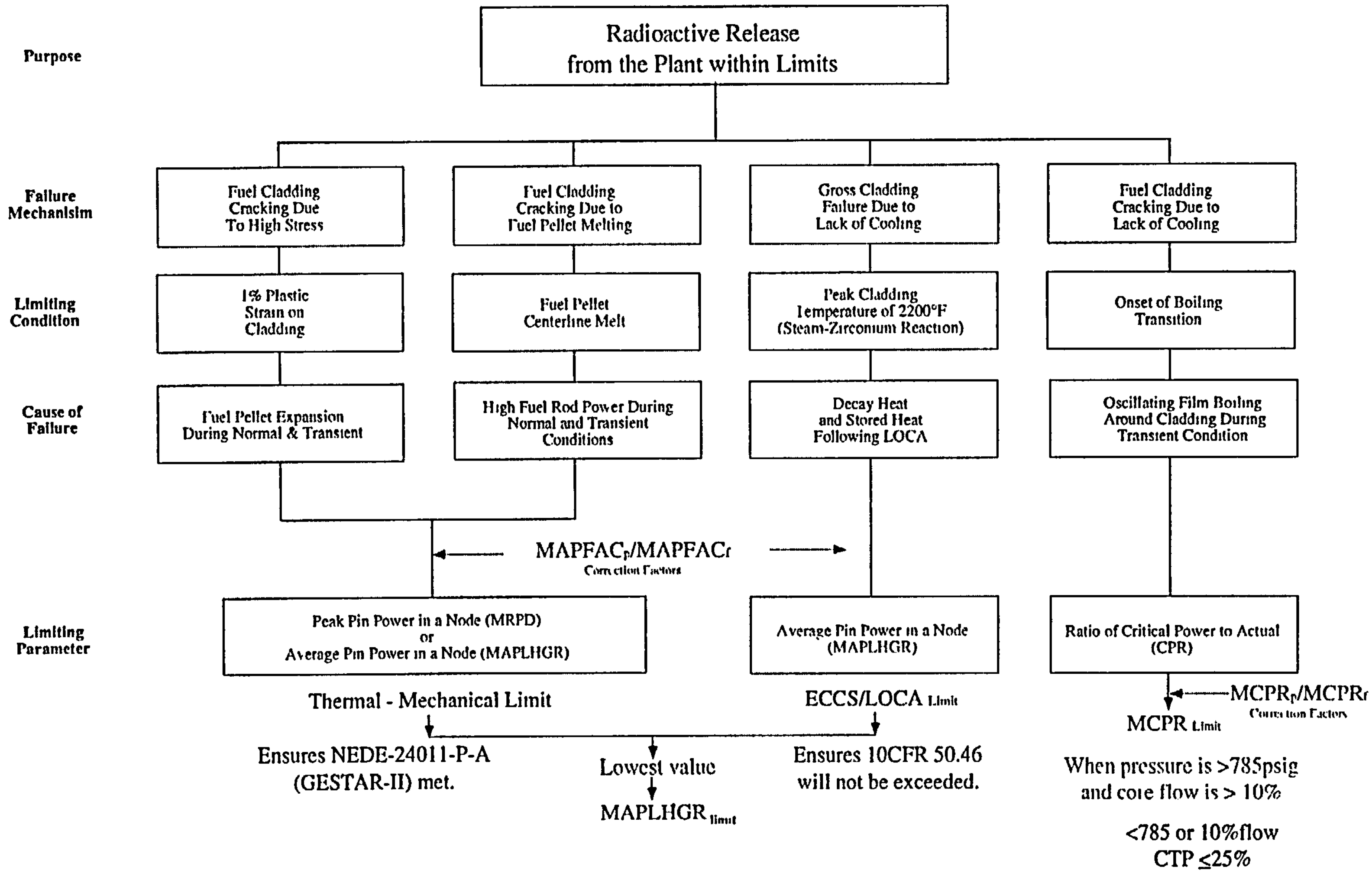


Figure 3.2-1 Thermal Limits

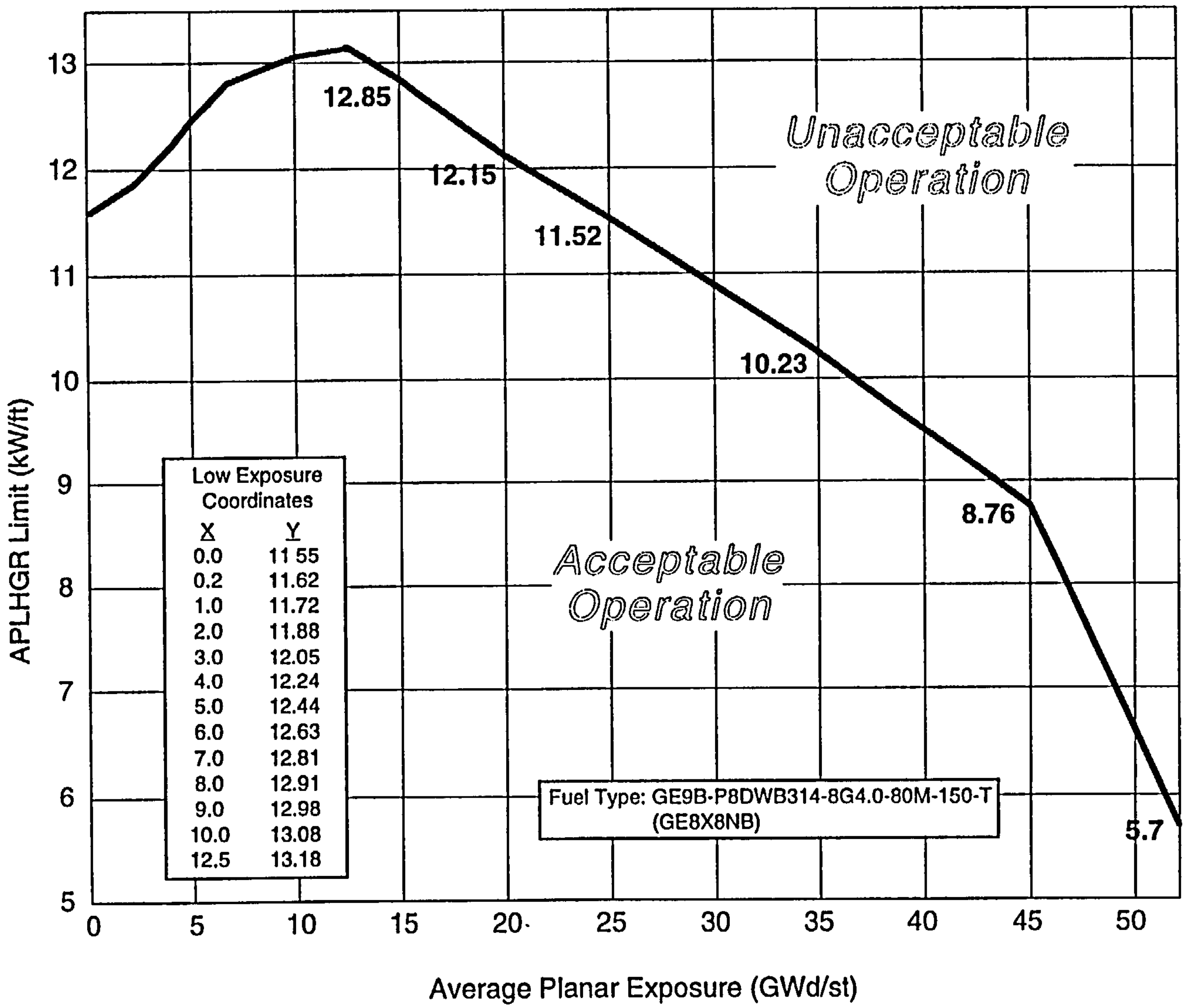
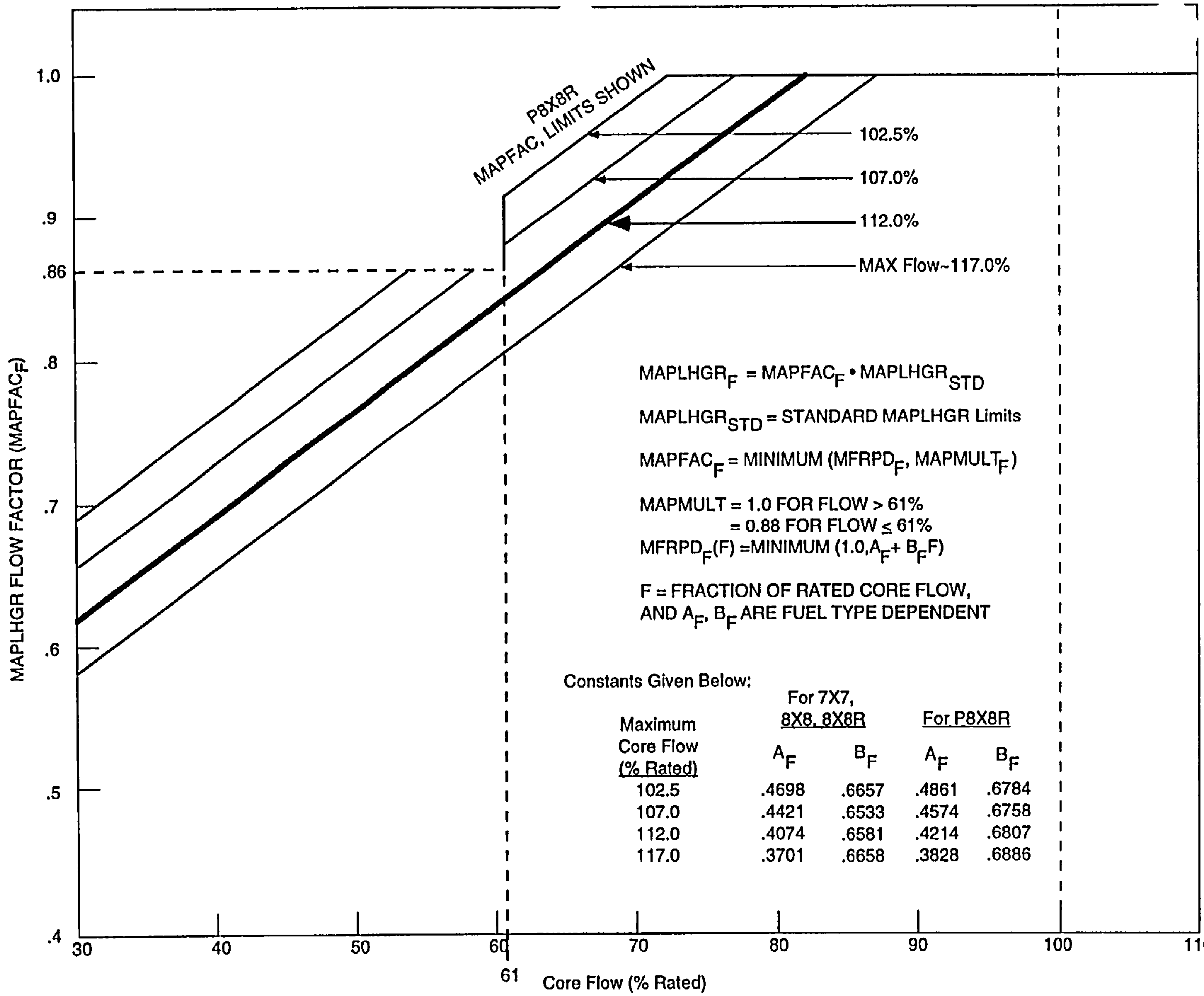


Figure 3.2-2 Average Planar Linear Heat Generation Rate Limit vs. Average Planar Exposure



$MAPLHGR_F = MAPFAC_F \bullet MAPLHGR_{STD}$

$MAPLHGR_{STD}$ = STANDARD MAPLHGR Limits

$MAPFAC_F$ = MINIMUM ($MFRPD_F$, $MAPMULT_F$)

MAPMULT = 1.0 FOR FLOW > 61%
= 0.88 FOR FLOW ≤ 61%

$MFRPD_F(F)$ =MINIMUM $(1.0, A_F + B_F F)$

F = FRACTION OF RATED CORE FLOW, AND A_F, B_F ARE FUEL TYPE DEPENDENT

Constants Given Below:

Maximum Core Flow (% Rated)	For 7X7, 8X8, 8X8R		For P8X8R	
	A_F	B_F	A_F	B_F
102.5	.4698	.6657	.4861	.6784
107.0	.4421	.6533	.4574	.6758
112.0	.4074	.6581	.4214	.6807
117.0	.3701	.6658	.3828	.6886

Figure 3.2-3 $MAPFAC_F$

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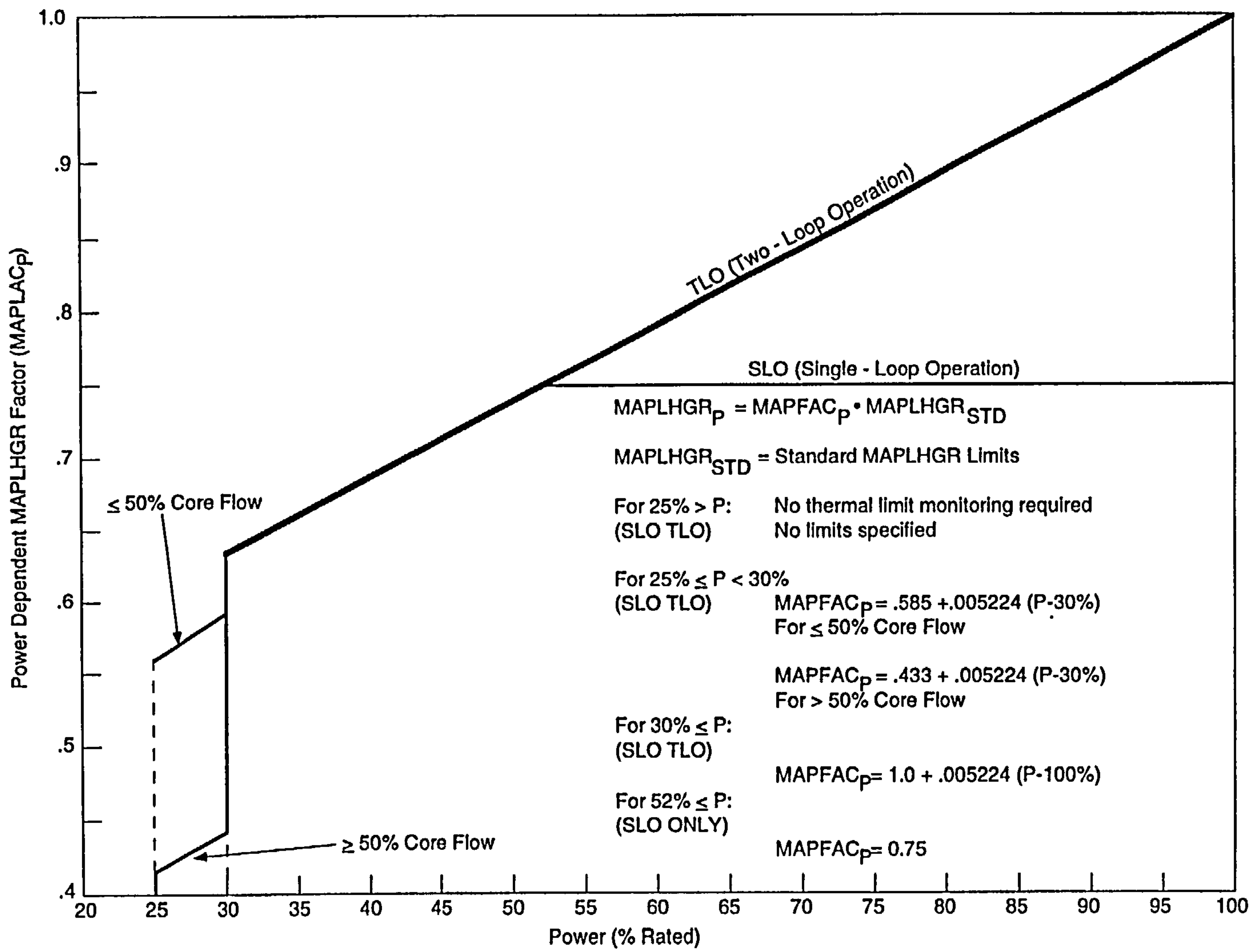

Figure 3.2-4 $MAPFAC_P$

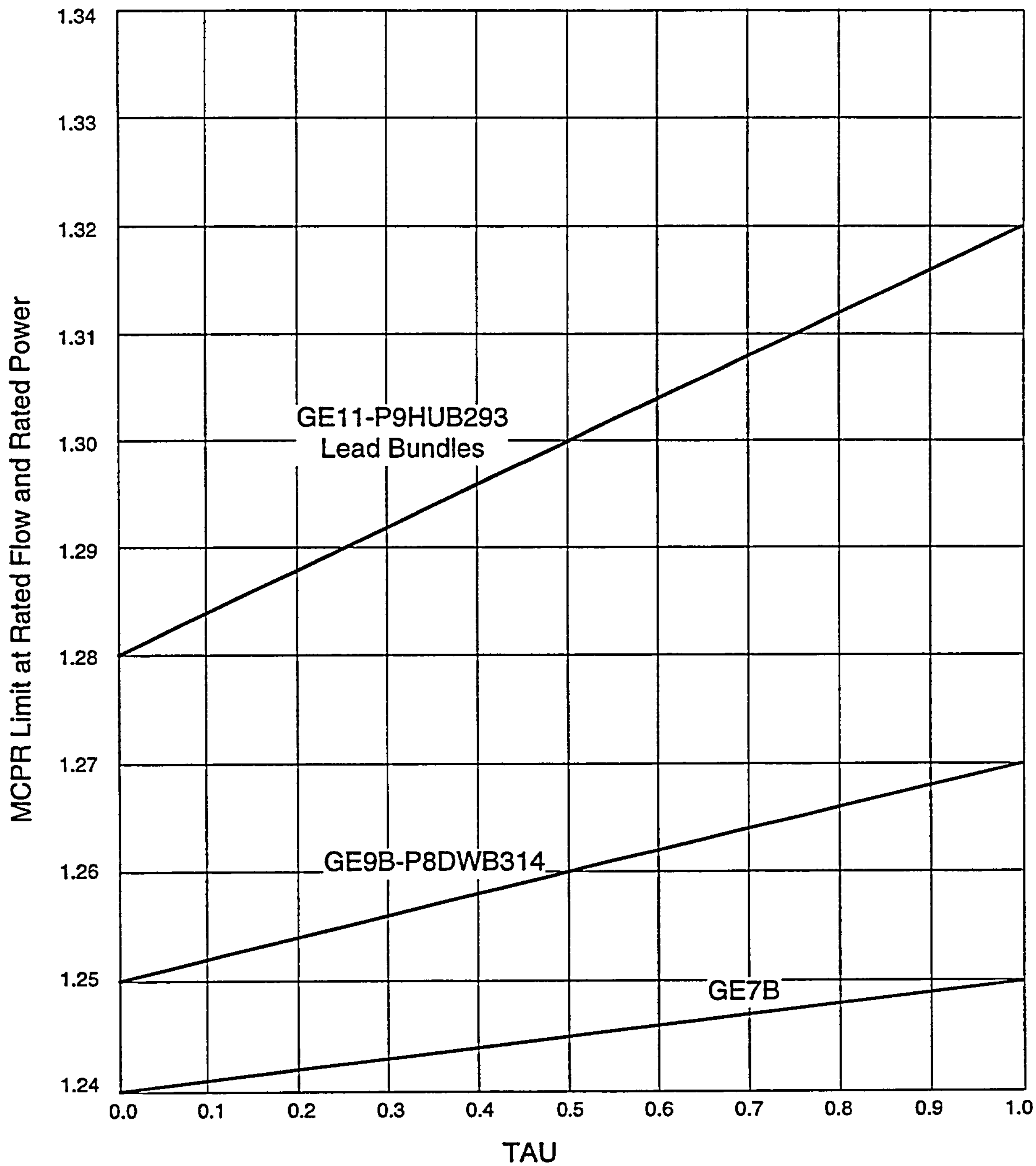

Figure 3.2-5 MCPR Limit as Function of Average Scram Time

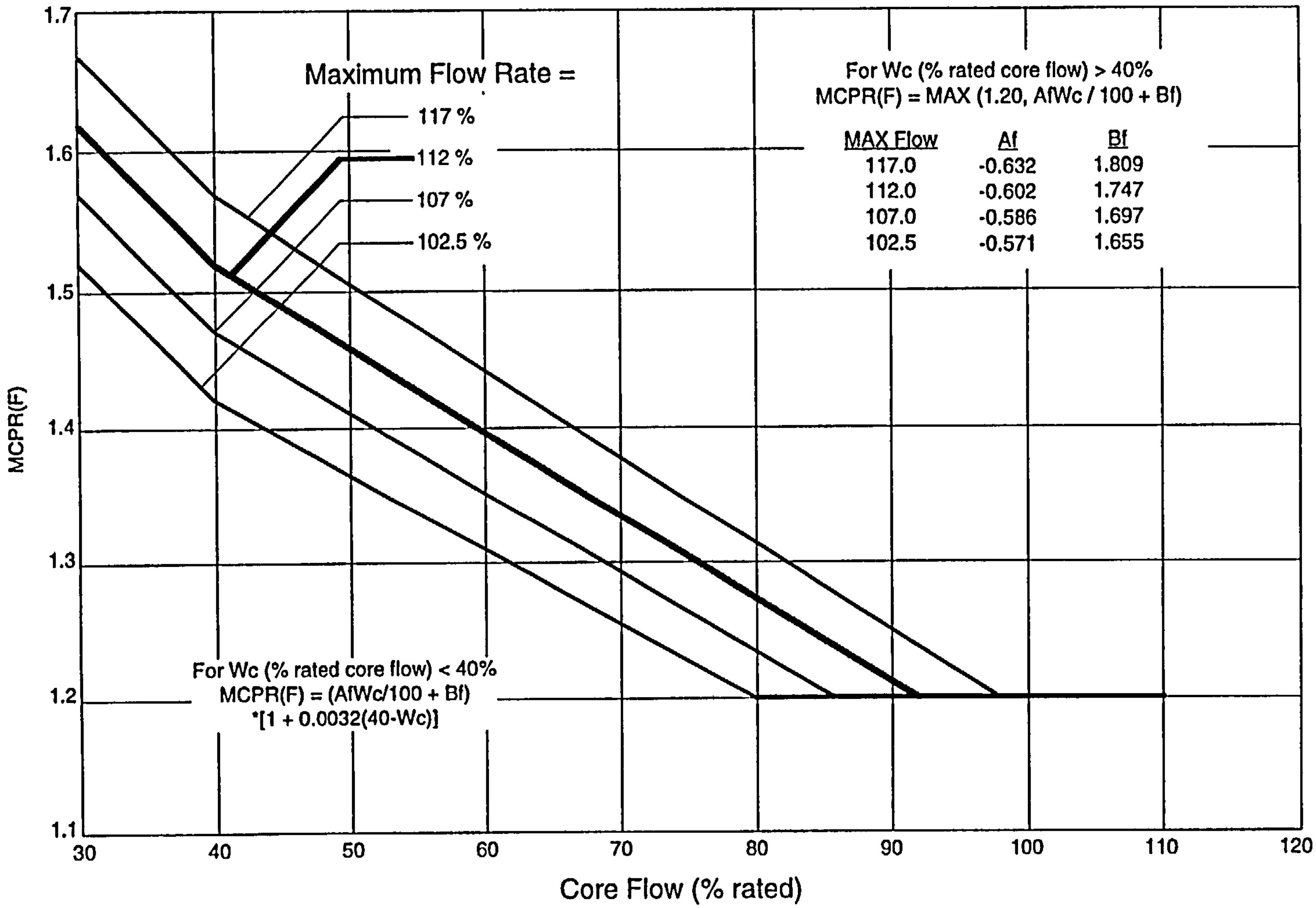

Figure 3.2-6 Flow - Dependent MCPR Limits, MCPR(F)

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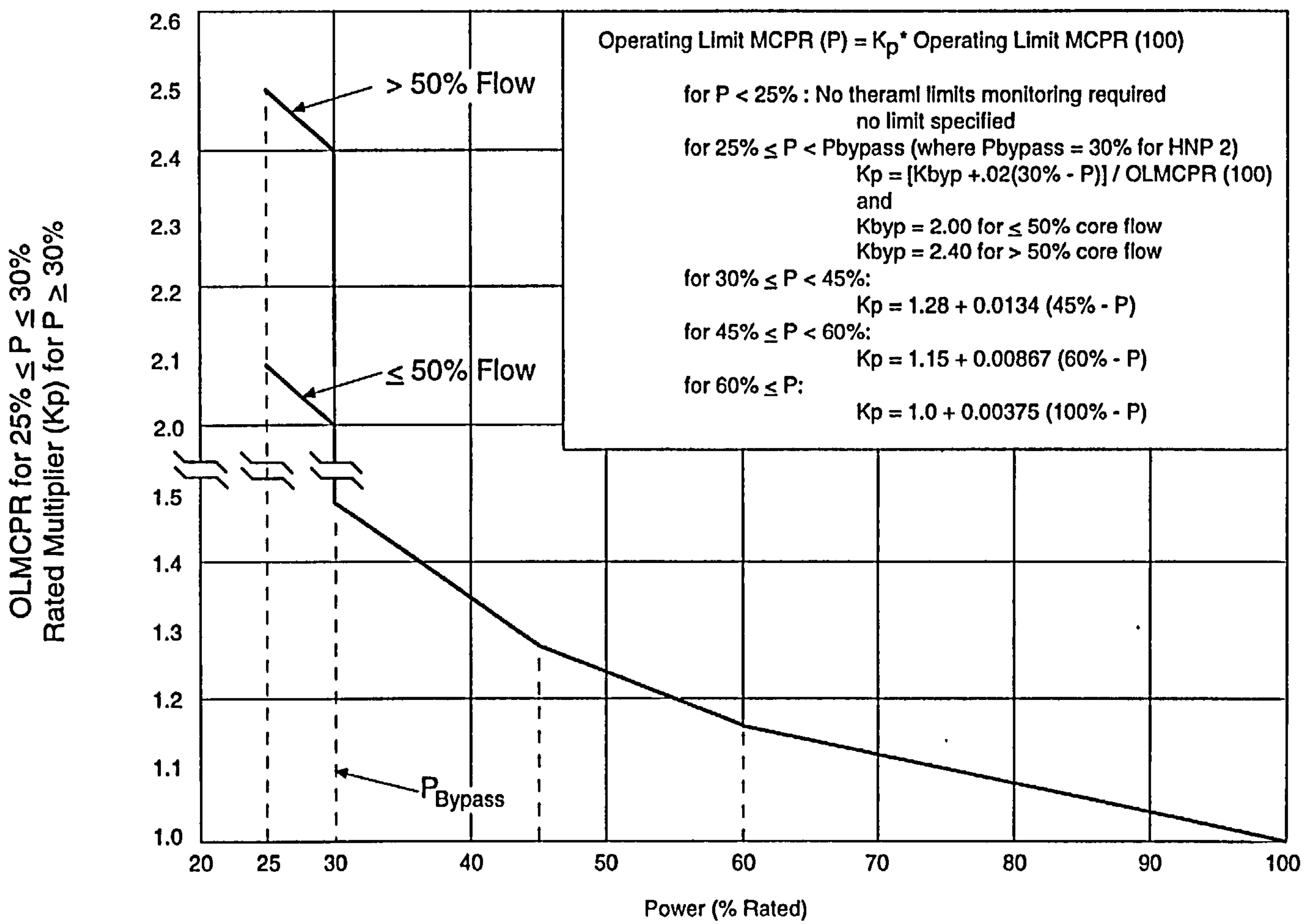

Figure 3.2-7 Power-Dependent MCPR Multiplier (Kp)

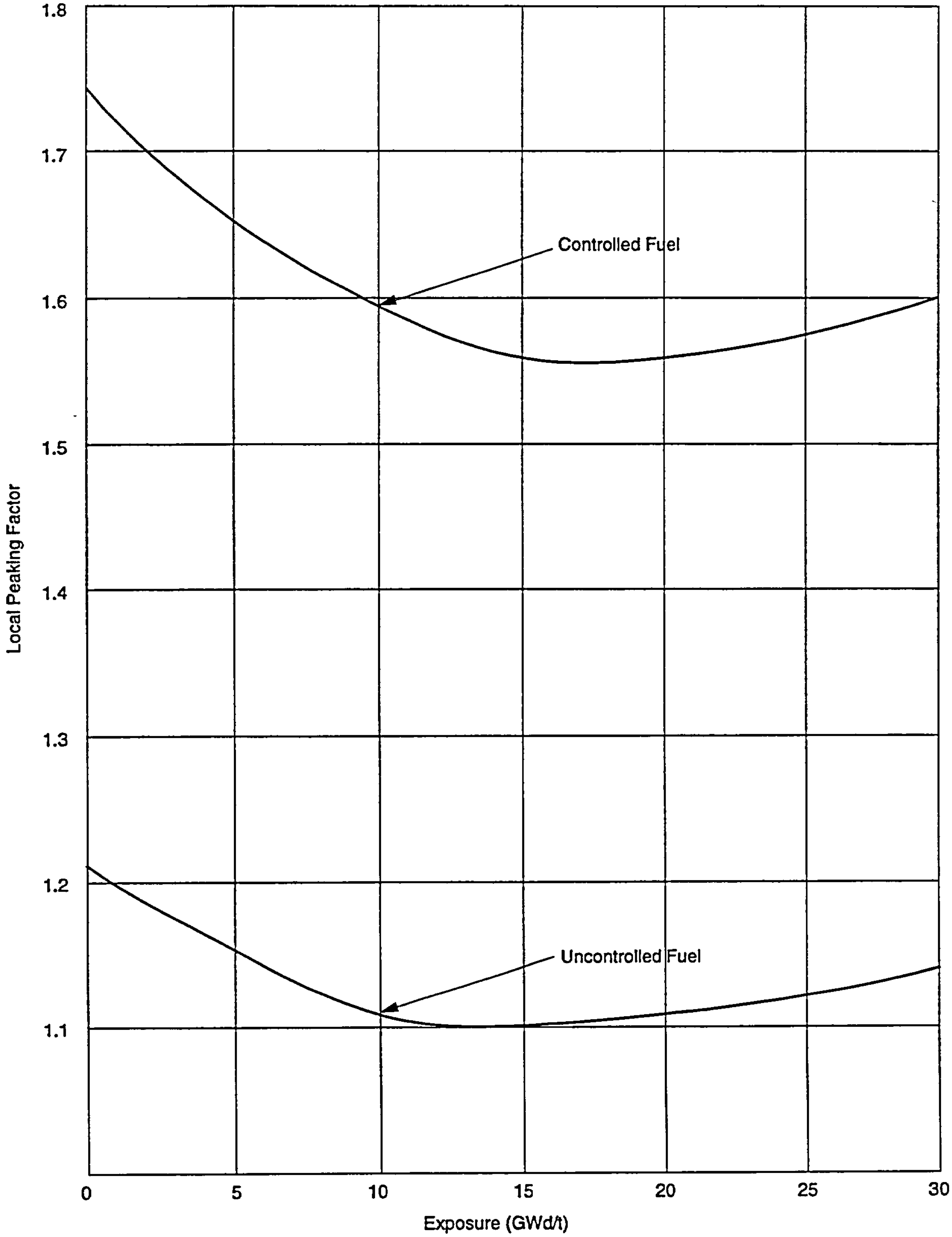

Figure 3.2-8 Typical Values for Local Peaking Factor

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3.3 CONTROL ROOM LOG 1

Learning Objectives:

1. Determine if any Technical Specification action statements are in effect.
2. Determine if any system addressed in the log is in an abnormal alignment.
3. Determine plant conditions relative to the instability region of the power/flow map.
4. Describe the basic method used to determine jet pump operability.
5. Describe the power condition with the most restrictive chloride limit.
6. Explain the need for PCIOMR restraints.
7. List the requirements for starting and operation of the recirculation pumps.

3.3.1 Introduction

Technical Specification chapter 3.3 consists of a typical control room log, Attachment A, that will require you to utilize Technical Specifications to address the learning objectives listed above.

3.3.2 Jet Pump Operability

It is important to verify that reactor operation is always consistent with the licensing basis. As part of the licensing basis, it assumes that the jet pumps are operating as designed because they contribute in the ability to re-flood the core to two-thirds core height and are a path for low pressure coolant injection flow into the reactor vessel (were applicable). Blockages in the recirculation loop would significantly decrease injection flow. Another important aspect is to recognize potential problems as soon as possible so as to minimize equipment damage and increase plant availability. Therefore, it is important to establish that the jet pumps are operable by monitoring their performance routinely.

The major instrumentation used for performance monitoring are the recirculation pump speed, recirculation pump flow, individual jet pump flow, jet pump loop flow, core flow and core plate differential pressure.

The principle method used is to compare actual conditions against expected conditions is daily record keeping. Such a method depends on instrument repeatability. This requires the accumulation of a "normal" data base for comparison to current operation. The most important part of this method is to *always use the same instrument used to obtain the data base.* This method also makes instrument calibrations critical.

Core flow versus square root core plate differential pressure, recirculation pump flow versus speed, jet pump flow versus recirculation pump speed and jet pump flow in differential pressure relationships are the most commonly used performance measures.

For illustrative purposes, the jet pump flow (differential pressure) relationship is discussed as a performance monitoring parameter. Individual jet pumps in a recirculation loop do not have the same flow. The unequal flow is due to:

- the drive flow manifold which does not distribute flow equally to all risers.
- individual jet pump manufacturing and installation tolerances.
- the flow resistance the jet pump encounters in the lower plenum and vessel annulus.

The flow (differential pressure) pattern or relationship of one jet pump to the loop mean is repeatable and is influenced by natural circulation at low core flow rates.

In addition, for constant drive flow, the jet pump inherently will not operate at a constant flow but will fluctuate over a flow range of about 5 percent. Further, due to the turbulence in the jet pump diffuser where the flow measurement pressure tap is located, the differential pressure signal is usually noisy when the jet pump is in operation. The constant motion of the individual jet pump flow indicators makes data acquisition difficult. However, the noise is the most positive indication that the jet pump is operating. A typical jet pump flow deviation relationship along with the acceptance criteria are shown in Figure 3.3-1.

3.3.3 Recirculation System

Operation with a reactor coolant recirculation loop inoperable is allowed, provided that adjustments to the flow reference scram and APRM rod block setpoints, MCPR cladding integrity safety limit, OLMCPR, and MAPLHGR limit are made. The adjustments to APLHGR and the MCPR limits that are required for single loop operation are provided in the Core Operating Limits Report. The flow reference simulated thermal power setpoint for single loop operation is reduced by the amount of $m\Delta W$, where m is the flow reference slope for the rod block monitor and ΔW is the largest difference between two loop and single loop effective drive flow when the active loop indicated flow is the same. This adjustment is necessary to preserve the original relationship between the scram trip and actual drive flow.

The possibility of experiencing limit cycle oscillations during single loop operation is precluded by restricting the core flow to greater than or equal to 45% of rated when core thermal power is greater than the 80% rod line.

3.3.3.1 Idle Recirculation Loop Startup

When restarting an idle pump, the discharge valve of the idle loop is required to remain closed until the speed of the faster pump is below 50% of its rated speed to provide assurance that when going from one to two loop operation, excessive vibration of the jet pump risers will not occur.

In order to prevent undue stress on the vessel nozzles and bottom head region the recirculation loop temperatures shall be within 50 ^{0}F of each other prior to startup of a an idle loop. Since the coolant in the bottom of the vessel is at a lower temperature than the water in the upper regions of the core, undue stress on the vessel would result if the temperature difference were greater than 145 ^{0}F. The loop temperature must be within 50 ^{0}F of the reactor vessel coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles.

3.3.4 PCIOMR

Preconditioning Interim Operating Management Recommendation (PCIOMR) discussion can be found in chapter 4.4 of this manual. Only a brief explanation of PCIOMR will be addressed here for the purpose of the control room log summary review.

PCIOMR is based on results of plant surveillance, fuel inspections, and individual fuel rod testing in the General Electric Test Reactor (GETR). Tests at GETR in 1971 and 1972 confirmed the mechanism and characteristics of the pellet clad interaction (PCI) failures observed in operating BWRs during rapid power increases. Beginning in 1972 test of production fuel rods demonstrated that a slow ascent to power would not only prevent failure, but that the slow ramp "preconditioned" the fuel to withstand subsequent rapid power changes at all levels up to that attained during the initial slow power increase.

For PCI to occur, both a chemical embrittling agent (fission products I and Cd) and high cladding stress are necessary. To eliminate the PCI problem General Electric introduced barrier-fuel. However, recent experiences at BWRs indicate that if a fault (crack) exists it will propagate very rapidly at high power. Therefore, General Electric has implemented a revised PCIOMR at plants with small fuel failures to prevent the zipper effect on the cladding.

The term "zipper effect" is the terminology initiated by General Electric to identify the rapid propagation of a fuel cladding failure pertaining to barrier fuel.

3.3.5 Chemistry

The water chemistry limits of the reactor coolant system are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low; thus the higher limit on chlorides is permitted during full power operation. During shutdown and refueling operations the temperature necessary for stress corrosion to occur is not present.

Conductivity measurements are required on a continuous basis since changes in this parameter is an indication of abnormal conditions. When the conductivity is within limits, the pH, chloride and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity meter inoperable, additional samples must be analyzed to ensure that the chlorides are not exceeding the limits.

3.3.6 Operational Leakage

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for unidentified leakage the probability is small that the imperfection of cracks associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the value specified or the leakage is located and known to be pressure boundary leakage the reactor will be shutdown to allow further investigation and corrective action.

3.3.7 Exercise

Attachment A represents a typical control room log at a BWR. With the aid of Technical Specifications and this text answer the learning objectives listed on page 3.3-1.

Attachment A
Control Room Log

NSS Name N. Jones		Shift No 2	Time 15:45	Date 04/2/98	
CRNSO R. Smith	603 Operator C. Gnffin	NASS T. Tym		Patrol NSO D. Cash	
Rx Mode 1	Rx Power 62%	MWe 525	Rx. Press 958	Rx Temp. 527	Rx Level 37
Recirc A Speed NA	Recirc B Speed 75%	Core Flow 45%	FW Flow 6.3	Stm. Flow 6.3	Drywell Press 0.11
SP Level 1.1	SP Temp. 79	MAPRAT 0.892	MFLPD 0.905	MFLCPR 0.874	

TIME	Comments
1600	Completed shift turnover: Recirculation pump 'A' was secured at 1400 for repairs on MG set.
1610	Performed PCIOMR ramp
1715	Placed 'B' RWCU F/D in service
1823	Attempted start of 'A' recirculation pump
1830	Noted 'B' RWCU outlet conductivity >0.1 µmho/cm
1845	Secured 'B' RWCU F/D , reactor water conductivity < 1 µmho/cm
2030	Started 'A' recirculation pump
2040	Placed recirculation system in master manual
2042	Received recirculation pump 'A' seal alarm
2053	Completed SI 4.4.3.2.1, leakage increased from 0.35 gpm to 1.42 gpm
2216	Performed PCIOMR ramp
2314	Called I &C to look at #10 jet pump due to abnormal oscillation indication

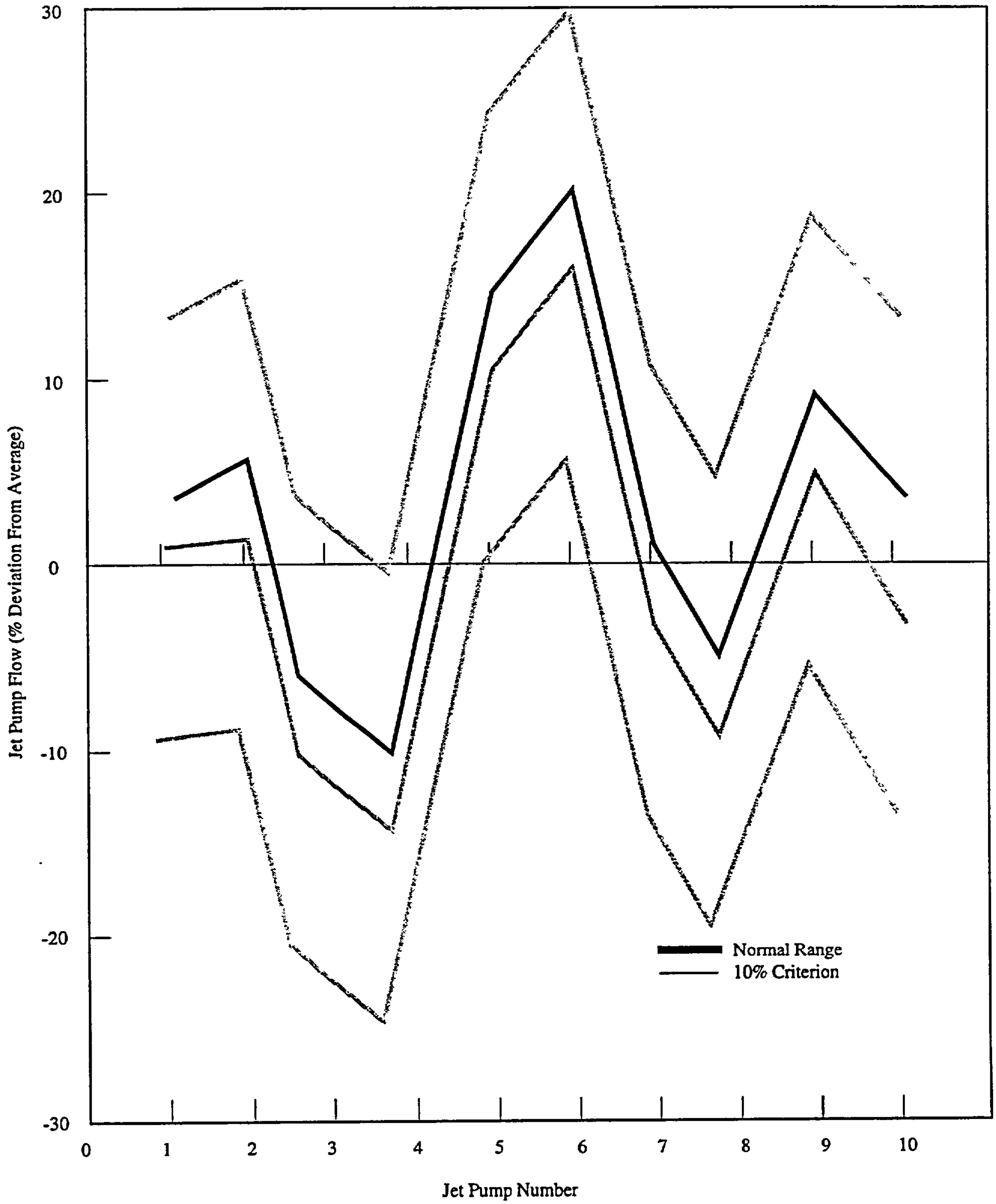

Figure 3.3-1 Jet Pump Flow Deviation

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3.4 CONTROL ROOM LOG 2

Learning Objectives:

1. Determine if any Technical Specification action statements are in effect.
2. Determine if any systems addressed in the log is in an abnormal alignment.
3. Determine plant status relative to power/flow map.
4. Explain the difference between an automatic isolation valve and a manual isolation valve.

3.4.1 Introduction

Technical Specification chapter 3.4 consists of a typical control room log, Attachment A, that will require you to utilize Technical Specifications to address the learning objectives listed above.

3.4.1 Pump and Valve Testing

Technical Specifications Section 5.5.6 specifies that inservice inspections of ASME Code Class 1, 2, and 3 components must be conducted. It also ensures that the pump and valves inspection will be performed in accordance with a periodically updated version of section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50, Section 50.55a. Exemptions from any of the above requirements has been approved in writing by the Commission and is not a part of these Technical Specifications.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout the Technical Specifications and to remove any ambiguities related to the frequencies for performing the required inservice inspection and testing activities.

3.4.2 High Pressure Coolant Injection

The High Pressure Coolant Injection (HPCI) system is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the nuclear system and loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCI system permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. HPCI system continues to operate until reactor vessel pressure is below the pressure at which LPCI system operation and Core Spray system operation can maintain core cooling. HPCI system capacity, 4250 gpm at 1135 and 165 psig, is selected to provide this required cooling.

With the HPCI system inoperable, adequate core cooling is assured by *operability* of the redundant and diversified Automatic Depressurization System (ADS) and the low pressure ECCSs. In addition, the Reactor Core Isolation Cooling (RCIC) System, system for which no credit is taken in the accident analysis, will automatically provide makeup at high reactor pressure. The Technical Specifications allowable out of service period of 14 days is based on the demonstrated operability of redundant and diversified low pressure ECCSs.

The surveillance requirements provide adequate assurance that the HPCI system will be *operable* when required.

3.4.3 Automatic Depressurization System

Upon failure of the HPCI system to function properly after a small break LOCA, the ADS automatically causes selected safety relief valves to open, depressurizing the reactor so that flow from the low pressure ECCSs can enter the core in time to limit fuel cladding temperature to less than 2200 ^{0}F. ADS is conservatively required to be operable whenever reactor pressure exceeds 150 psig even though low pressure ECCSs provide adequate core cooling up to 350 psig.

ADS automatically controls seven selected safety-relief valves although the accident analysis only takes credit for six valves. It is therefore appropriate to permit one valve to be out-of-service for 14 days without materially reducing system reliability.

ADS accumulators are sized such that, following loss of the pneumatic supply, at least two valve actuation will be possible with the drywell at 70% of its design pressure. The allowable accumulator leakage criterion ensures the above capability for 30 minutes following loss of the pneumatic supply.

3.4.4 Reactor Core Isolation Cooling System

The Reactor Core Isolation Cooling (RCIC) system is provided to assure adequate core cooling in the even of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without requiring actuation of any of the ECCSs. The RCIC system is conservatively required to be operable whenever reactor pressure exceeds 150 psig even though the RHR system provides adequate core cooling up to 350 psig.

The RCIC system specifications are applicable during CONDITIONS 1, 2, and 3 when reactor pressure exceeds 150 psig because RCIC is the primary non-ECCS source of emergency core cooling when the reactor is pressurized.

Two sources of water are available to the RCIC system. Suction is initially taken from the condensate storage tank and is automatically transferred to the suppression pool upon low CST level or high suppression pool level.

With RCIC inoperable, adequate core cooling is assured by the demonstrated operability of the HPCI system and justifies the specified 14 day out-of-service period.

3.4.5 RHR System -Low Pressure Coolant Injection

The LPCI mode of the Residual Heat Removal System is provided to assure that the core is adequately cooled following a LOCA. Two subsystems, each with two pumps, provide adequate core flooding for all break sizes from 0.2 ft^2 up to and including the double-ended reactor recirculation line break, and for small breaks following depressurization. LPCI system specifications are applicable during conditions 1, 2 and 3 because LPCI is a primary source of water for flooding the core after the reactor vessel is depressurized.

When in conditions 1, 2, or 3 with one LPCI pump inoperable or one LPCI subsystem inoperable, adequate core flooding is assured by the operability of the redundant LPCI pumps or subsystems and both CS subsystems. The reduced redundancy justifies the specified 7 day out-of-service period.

Shutdown Cooling

Irradiated fuel in the shutdown reactor core generates heat during the decay of fission products which increases the temperature of the reactor coolant. This decay heat must be removed to reduce the temperature ≤212°F. This decay heat removal is in preparation for performing refueling or maintenance operations, or for keeping the reactor in the Hot Shutdown condition.

The two redundent, manually controlled shutdown cooling subsystems of the RHR system provide decay heat removal. Each loop consists of two motor driven pumps, a heat exchanger, and associated piping and valves. Both loops have a common suction from the same recirculation loop. Each pump discharges the reactor coolant, after circulation through the respective heat exchanger, to the reactor via the associated recirculation loop. Two RHR shutdown cooling subsystems are required to be operable, and when no recirculation pump is in operation, one shutdown cooling subsystem must be in operation. An operable RHR shutdown cooling subsystem consists of one operable pump and associated heat exchanger, piping and valves which can provide the capability to reduce and maintain temperature <212°F.

3.4.6 Suppression Chamber

The operability of the suppression chamber in conditions 1, 2, or 3 is required by specification 3.6.2.1 of Technical Specifications. The suppression pool is designed to absorb the decay heat and sensible energy released during a reactor blowdown from safety/relief valve discharges for from design basis accidents. The suppression pool must quench all the steam released through the downcomer lines during a loss of coolant accident. This is the essential mitigative feature of pressure suppression containment that ensures that the peak containment pressure is maintained below the maximum allowable pressure for DBAs. The suppression pool must also condense steam from steam exhaust lines in the HPCI and RCIC systems. Technical concerns that lead to the development of suppression pool average temperature limits are :

a. Complete steam condensation;
b. Primary containment peak pressure and temperature;
c. Condensation oscillation loads; and
d. Chugging loads.

The postulated DBA against which the primary containment performance is evaluated is the entire spectrum of postulated pipe breaks within the primary containment. Inputs to the safety analyses initial suppression pool temperature and water volume. An initial pool temperature of 110°F is assumed for analyses. Reactor shutdown at a pool temperature of 110°F and vessel depression at a pool temperature of 120°F are assumed. The limit of 105°F, at which testing is terminated, is not used in the safety analyses because DBAs are assumed to not initiate during unit testing.

Average temperature ≤100°F when any operable intermediate range monitor channel is > 25/40 divisions of full scale on range 7 and no testing that adds heat to the suppression pool is being performed. This requirement ensures that licensing bases initial conditions are met.

Average temperature ≤105°F when any operable intermediate range monitor channel is > 25/40 divisions of full scale on range 7 and testing that adds heat to the suppression pool is being performed. This required value ensures that the unit has testing flexibility, and was selected to provide margin below 110°F limit at which reactor shutdown is required.

Average temperature ≤105°F when any operable intermediate range monitor channel is > 25/40 divisions of full scale on range 7. This requirement ensures that the unit will be shutdown at > 110°F. Note that 25/40 divisions of full scale on IRM range 7 is a convenient measure of when the reactor is producing power essentially equivalent to 1% power.

Repair work might require making the suppression chamber inoperable. Therefore it is permitted to drain the suppression pool in condition 5.

The suppression chamber water provides the heat sink for the reactor coolant system energy release. The suppression pool volume ranges between approximately 86,000 ft^3 at the low water level limit of 146 inches and approximately 90,000ft^3 at the high water level limit of 150 inches. If the suppression pool water level is to low, an insufficient amount of water would be available to adequately condense the steam from S/RV quenchers, main vents, or HPCI and RCIC turbine exhaust limes. Low suppression pool water level could also result in an inadequate emergency makeup water source to the ECCSs. The lower volume would also absorb less steam energy before heating up excessively.

If the suppression pool water level is too high, it could result in insufficient volume to accommodate non-condensable gases and excessive pool swell loads during a DBA LOCA. Therefore, a maximum pool water level is specified.

3.4.7 Primary Containment Isolation Valves

The operability of the primary containment isolation valves ensures that the primary containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the primary containment atmosphere or pressurization of the containment. Primary containment isolation within the time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA. Automatic isolation valves are valves that receive automatic signals either from isolation logic or system operational signals. Manual containment isolation valves are valves that receive no automatic closure signals and are closed either remotely from the control room or locally at the valve.

3.4.8 Line Communication with Primary Containment

Several lines which penetrate the containment and communicate with its atmosphere are provided with isolation valves outside primary containment, rather than one isolation valve inside and one isolation valve outside primary containment. This deviation from GDC is considered safe and adequate because:

A. Lines which penetrate the containment for atmosphere sampling or processing terminate at the inboard end of the weld, within the drywell penetration sleeve. The sleeve and the piping connect to it on Seismic Category I, Quality Group B up to and including at least the second primary containment isolation valve.

   Installation of the inboard isolation valve inside primary containment would require supporting the valves from the drywell shell, resulting in additional welds, and/or extending the piping, and adding supports from other structural members within the drywell. Since these valves inside would severely impede accessibility for inspection and maintenance of the valves and other equipment.

B. Placing the valves inside the containment would subject them to an inimical environment and, thus, increase the probability of failure.

The environment within the drywell and suppression chamber post-LOCA could be especially detrimental to the operation of the drywell and wetwell spray valves, since these valves would be required to function during the postulated containment pressure transient. The design spray coverage further necessitates the location of the spray header as close as practical to the interior of the drywell and wetwell shells.

Therefore, the isolation calves for each spray header is installed outside of primary containment. The outboard barrier is the closed RHR system. In addition to the two barriers required by GDC 56, the wetwell and drywell spray lines each have a motor operated valve installed inboard of the containment isolation valves. This design reflects the importance of avoiding an in advertent initiation of containment sprays during plant operation. These valves also contribute additional conservatism to the containment isolation provisions since they shut, if open upon receipt of the LOCA initiation signal.

C. Valves are accessible in systems which must be available for long-term operation following an accident.

D. Isolating valves installed outside primary containment are compatible with minimizing personnel exposure during maintenance and inspections. Isolation valves for this category of line are either locked closed, administratively close, or are automatically closed upon receipt of an isolation signal.

The isolation valves in each line are installed as close together and as close to the primary containment as practical.

Influent Lines to Suppression Pool

The reasons for not placing valves inside the suppression chamber (pool) are similar to those already mentioned. The following discussion provides unique considerations as to the types of valves and isolation capabilities:

The RCIC and HPCI turbine exhaust lines, HPCI turbine condensate line, and RCIC vacuum pump discharge line.

These line penetrate the wetwell and discharge below the minimum water level. Two primary containment isolation valves are provided outside the wetwell on each line. The inboard isolation valve for each line is a motor operated locked open globe stop check valve. When in its normal position, open, the valve allows flow into the suppression pool. The valve may be manually closed, from the control room, for long term leakage control. The outboard valve is a simple swing check valve and functions as a redundant isolation valve to ensure backflow from the suppression pool is prohibited.

Since HPCI and RCIC are ESF systems, check valves are used as isolation valves to optimize system operability.

Minimum Flow and Test Lines

These lines have isolation capabilities which are commensurate with the importance to safety of isolating these lines. The HCPI and RCIC minimum flow lines have two valves in series, both located outside containment. The RHR and CS minimum flow lines also have two isolation valves. One isolation valve is motor operated and the other is a swing check valve.

The core spray test line has a single automatic isolation valve installed outside primary containment. The core spray system is a closed system therefore, the system is the second isolation barrier and no further isolating is required. The residual heat removal system test line is has a single automatic isolation valve outside primary containment and also is a closed system, therefore it too requires no other isolation valves.

Effluent Lines from Suppression Pool

It should be noted that GDC 56 does not reflect consideration of the BWR containment design. Certain lines, such as the RHR, CS, HPCI and RCIC suction lines, penetrate below the water line and therefore, do not communicate with the containment atmosphere. These lines do have an isolation valve located inside containment, under water. This would result in introducing a potentially unreliable valve in a highly reliable system, thereby compromising design. For this reason, these line incorporate isolation valves outside the containment.

3.4.9 Exercise

Attachment A represents a typical control room log at a BWR/4. With the aid of Technical Specifications and this text, answer the learning objectives on page 3.4.1.

3.4.10 Additional differences/problems with other technical specifications.

Suppression pool level and temperature limits are found in different section in the standard/custom technical specifications. In Section 3/4.5, emergency core cooling systems, under depressurization systems suppression chamber the minimum and maximum water level limits are found. In order to find the water temperature limits you will have to go to section 3/4.6, containment systems, under suppression chamber. In addition, the containment isolation valves are listed in technical specifications section 3/4.6.3, primary containment isolation valves. In this section the automatic and manual isolation valves are found. Remembering that automatic isolation valves only means that the valves receive an automatic signal to close, which could be a system signal or to an isolation signal.

Manual containment isolation valves are valves that receive no automatic closure signals and are closed either remotely from the control room or locally at the valve.

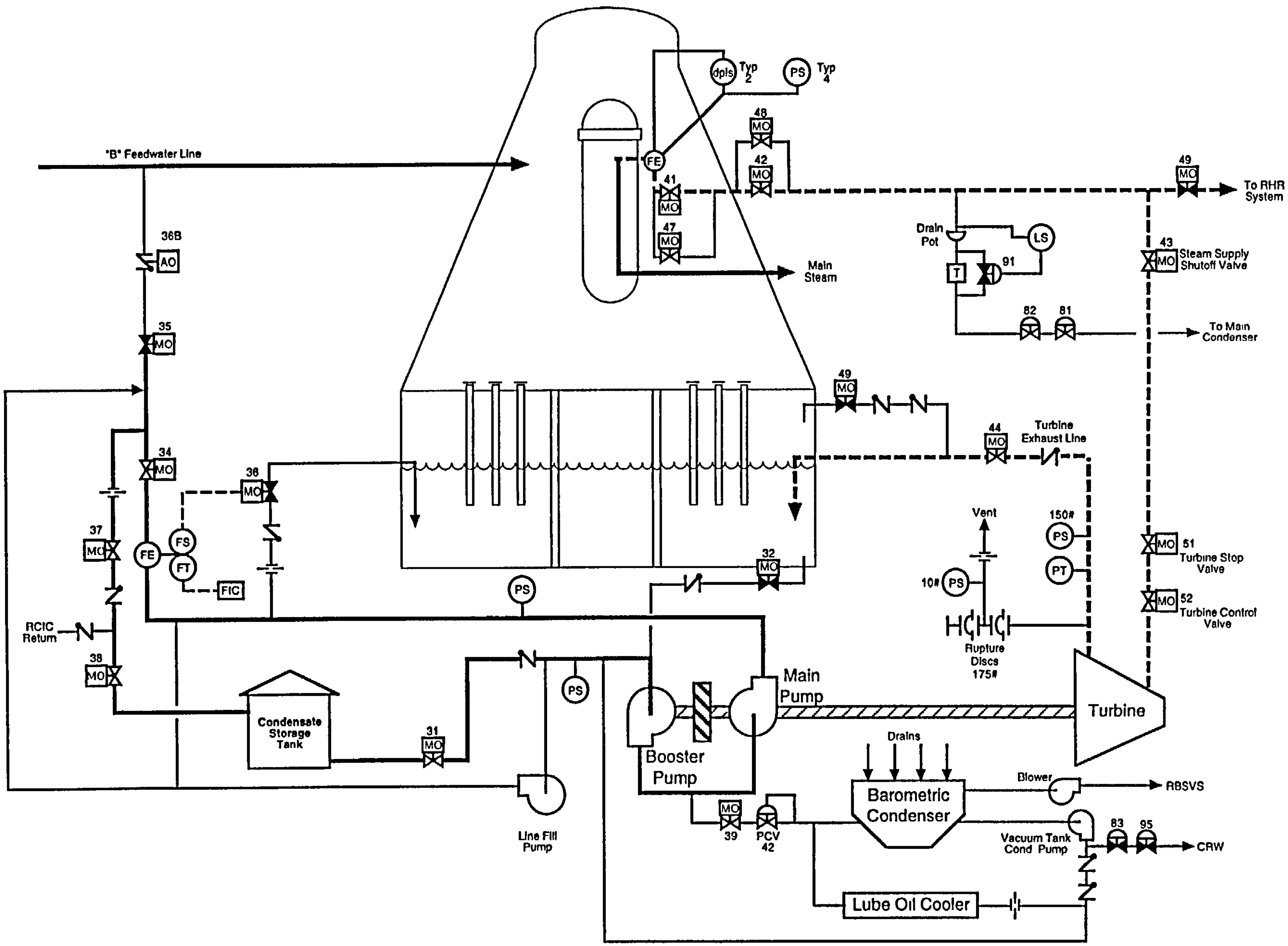

Figure 3.4-1 High Pressrue Coolant Injection System

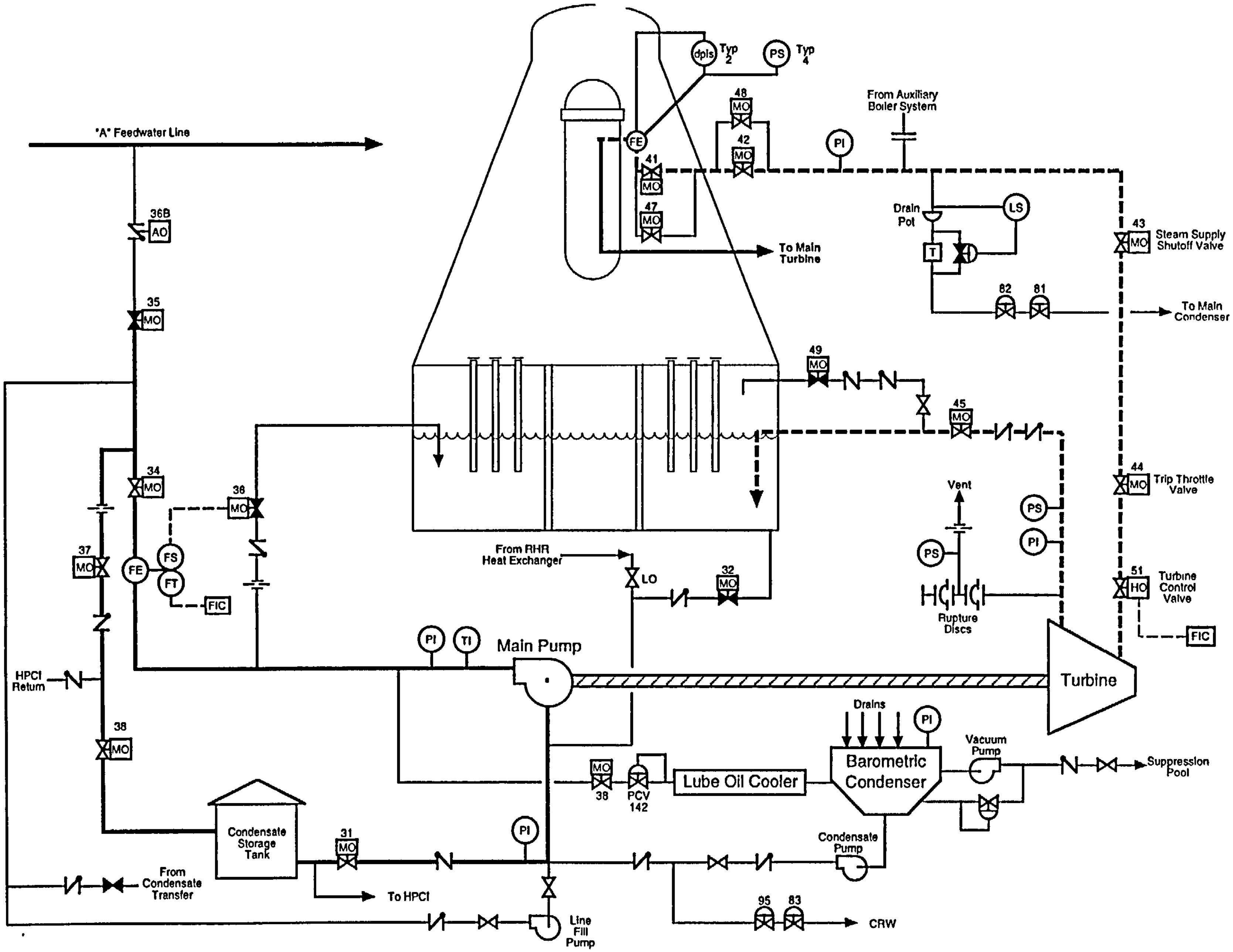

Figure 3.4-2 Reactor Core Isolation System

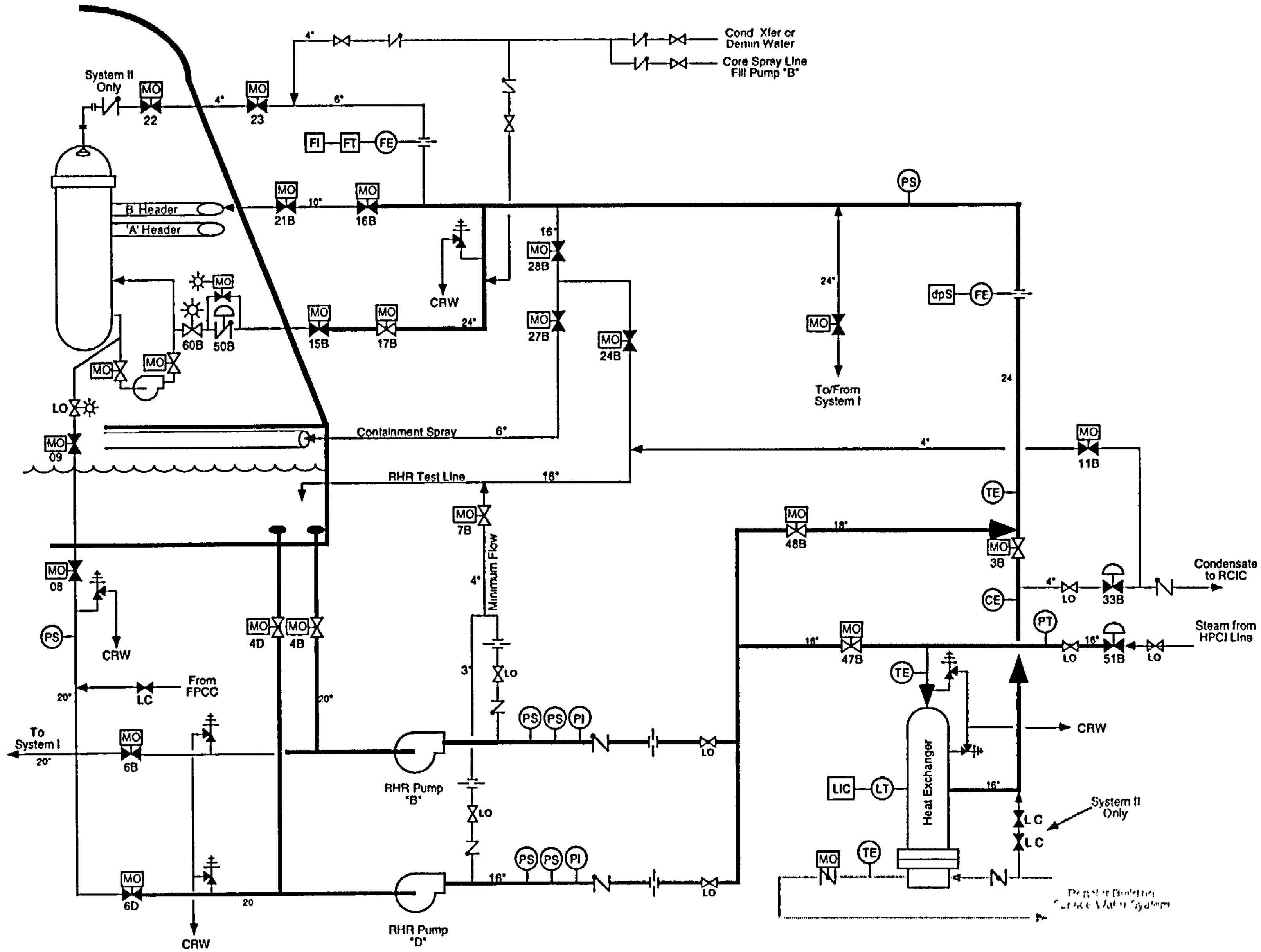

Figure 3.4-3 RHR System II

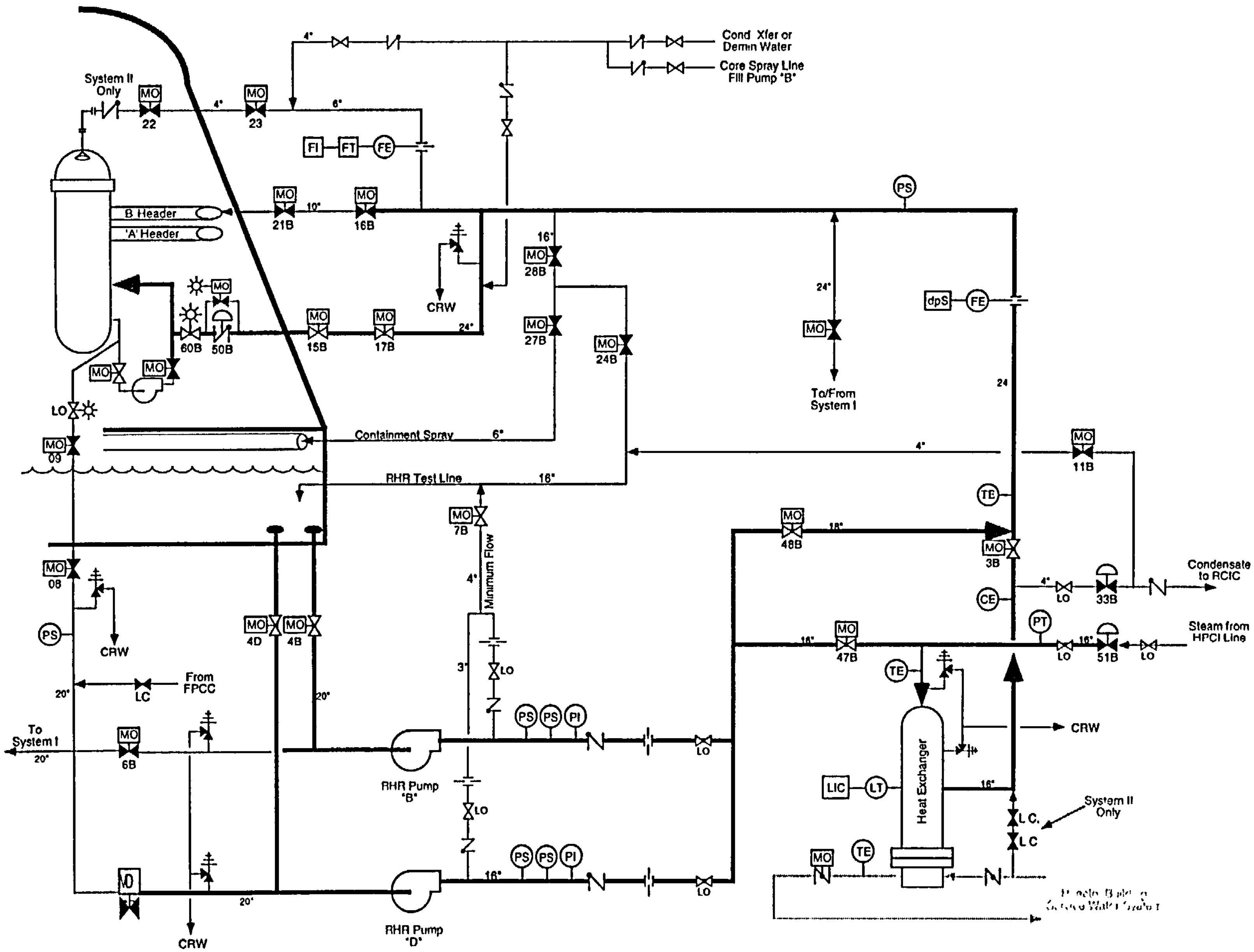

Figure 3.4-3a RHR LPCI

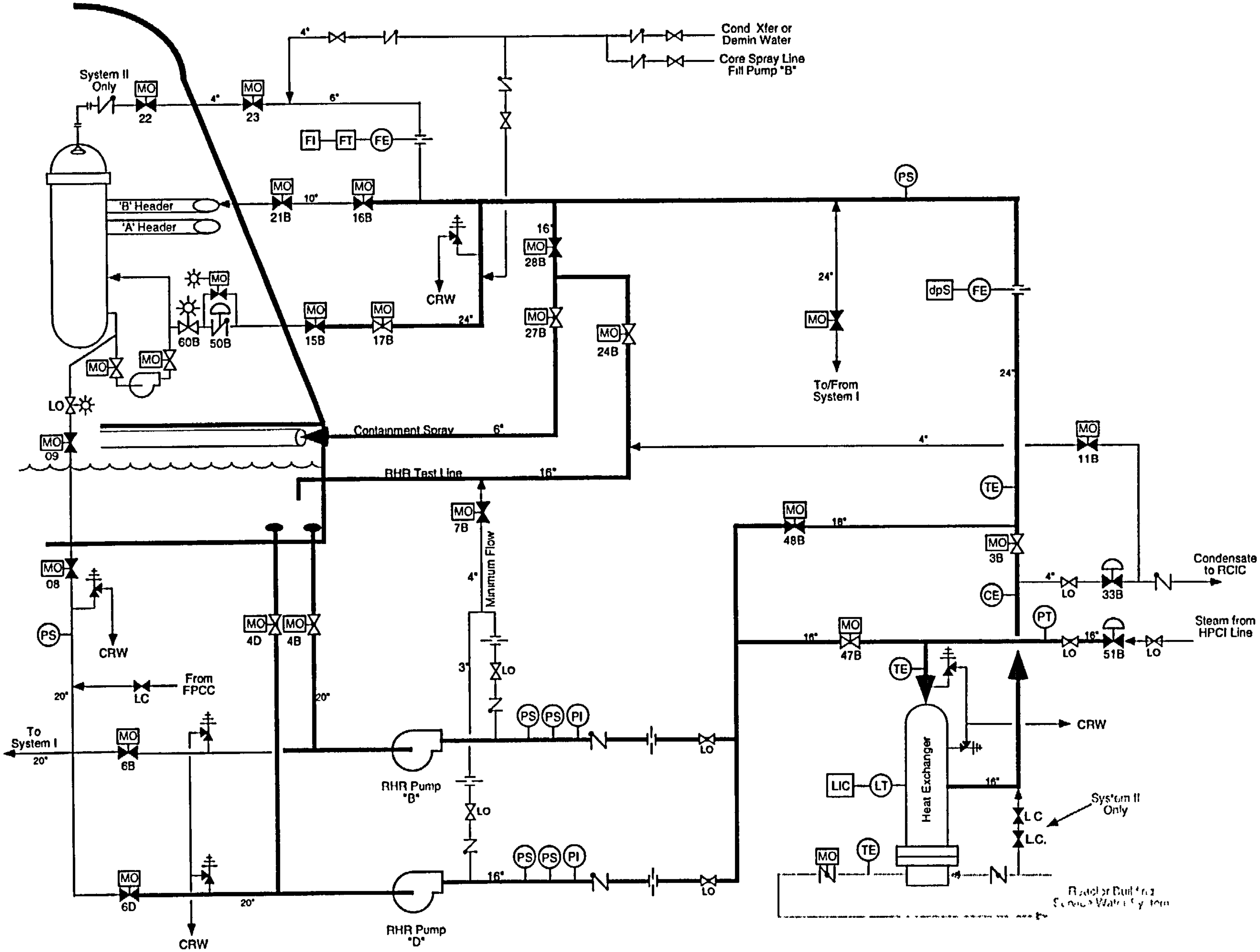

Figure 3.4-3b RHR SP Cooling

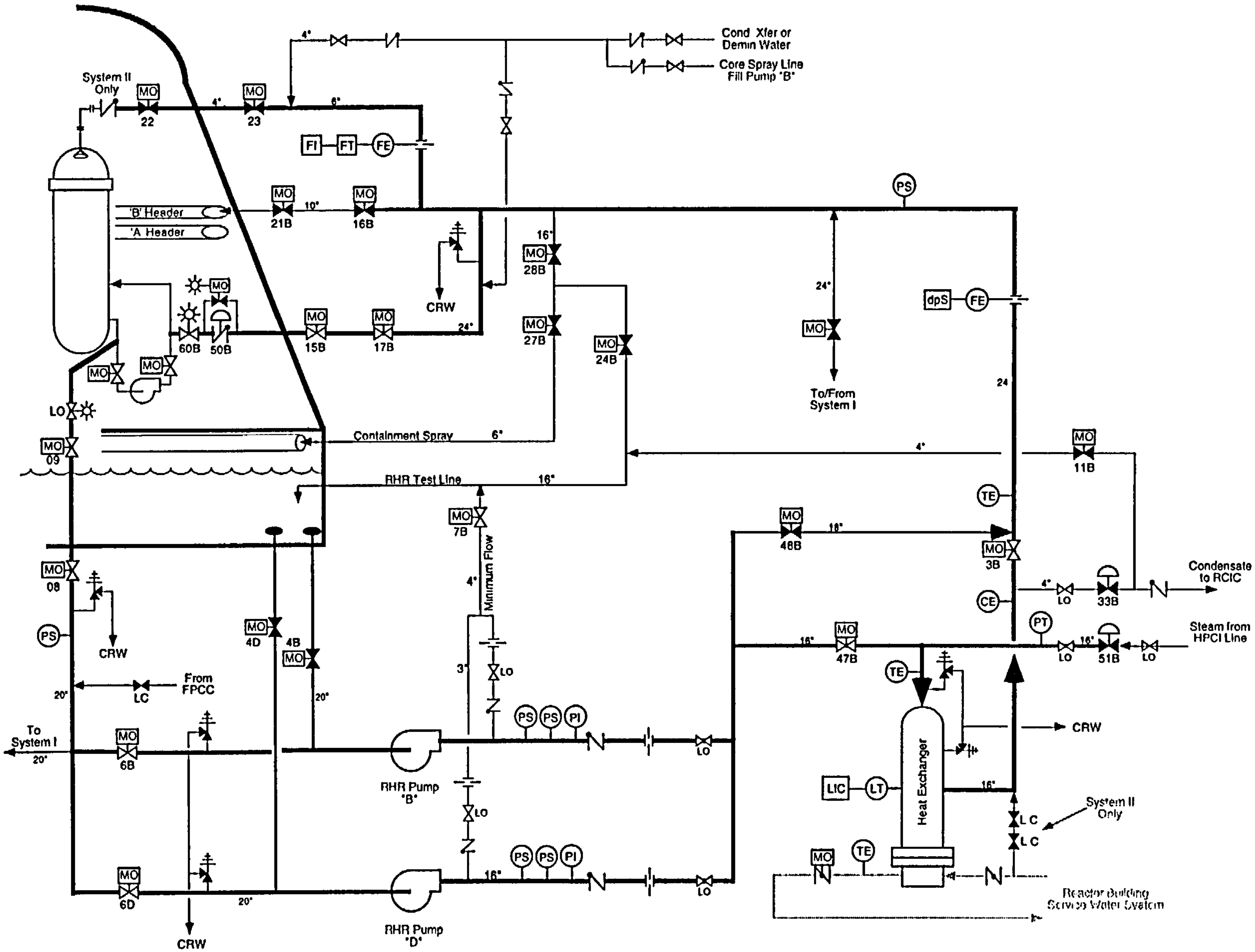

Figure 3.4-3c RHR SD Cooling

0295

Attachment A
Control Room Log

NSS Name	N. Jones	Sft No.	2	Time	15:45	Date	04/2/98				
CR NSO	R. Smith	603 Operator	C. Griffin	NASS	T. Tym	Patrol NSO	D. Cash				
Rx Mode	1	Rx. Power	90%	MWe	525	Rx. Press	958	Rx. Temp	527	Rx Level	37
Recirc A Speed	77%	Recirc B Speed	77%	Core Flow	100%	FW Flow	6.3	Stm Flow	6.3	Drywell Press	0.11
SP Level	1.1	SP Temp	79	MAPRAT	0.892	MFLPD	0.905	MFLCPR	0.874		

TIME	Comments
2349	Relieved the watch. verified plant status and equipment out of service correct.
0125	Started HPCI flow rate surveillance.
0225	Completed HPCI surveillance satisfactory.
0230	Started RCIC flow rate surveillance.
0245	Completed RCIC surveillance satisfactory.
0250	Placed RHR division 1 in sppression pool cooling mode. Suppression pool temperature 103 °F.
0323	Started Valve operability of RHR division 2.
0333	Terminated RHR valve operability . The D RHR pump suppression pool suction valve would not open from the control room. Opened the valve locally with hand wheel, to ensure system operability.

TABLE T7.0-1 (Sheet 1 of 31)
PRIMARY CONTAINMENT ISOLATION DEVICES - TESTABLE PENETRATIONS

PEN. NO.	DESCRIPTION	BARRIER ORIENT.	BARRIER TYPE (MPL)	VALVE TYPE (A)	TEST TYPE (B)	POWER TO OPEN (A), (C)	POWER TO CLOSE (A), (C)	ISOL. GP. (D)	MAX. OPER. TIME (sec)	NORM. POS.	POS. ON ISOL. (E)	NOTES (F)
1A	Equipment Hatch	Inboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5,54
1A	Equipment Hatch	Outboard	-	-	-	-	-	-	-	-	-	
1B	Equipment Hatch	Inboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5,54
1B	Equipment Hatch	Outboard	-	-	-	-	-	-	-	-	-	
2	Personnel Airlock	Inboard	Double O-Ring	-	B	-	-	-	-	-	-	1,3,6,54
2	Personnel Airlock	Inboard		-		-	-	-	-	-	-	
2	Personnel Airlock	Outboard	Double O-Ring	-	B	-	-	-	-	-	-	1,3,6,54
2	Personnel Airlock	Outboard	Barrel	-	B	-	-	-	-	-	-	55
3	H202 Sample Supply	Inboard	2P33-F002	AO Globe	C	Spring	Air/DC	10	5	Open	Closed	1,2,3,4
3	H202 Sample Supply	Outboard	2P33-F010	AO Globe	C	Spring	Air/AC	10	5	Open	Closed	1,2,3,4
4	Drywell Head Access Hatch	Inboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5,54
4	Drywell Head Access Hatch	Outboard	-	-		-	-	-	-	-	-	
5A	Drywell to Torus Vent Line	Inboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5,14
5A	Drywell to Torus Vent Line	Outboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5,14
5B	Drywell to Torus Vent Line	Inboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5,14
5B	Drywell to Torus Vent Line	Outboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5,14
5C	Drywell to Torus Vent Line	Inboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5,14
5C	Drywell to Torus Vent Line	Outboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5,14
5D	Drywell to Torus Vent Line	Inboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5,14
5D	Drywell to Torus Vent Line	Outboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5,14
5E	Drywell to Torus Vent Line	Inboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5,14
5E	Drywell to Torus Vent Line	Outboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5,14
5F	Drywell to Torus Vent Line	Inboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5,14
5F	Drywell to Torus Vent Line	Outboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5,14
5G	Drywell to Torus Vent Line	Inboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5,14
5G	Drywell to Torus Vent Line	Outboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5,14
5H	Drywell to Torus Vent Line	Inboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5,14
5H	Drywell to Torus Vent Line	Outboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5,14
6	CRD Removal Hatch	Inboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5,54
6	CRD Removal Hatch	Outboard	-	-		-	-	-	-	-	-	
7A	Main Steam	Inboard	2B21-F022A	AO Globe	C	N2/AC/DC	N2/Spring	1	3<T<5	Open	Closed	1,4,7,27
7A	Main Steam	Inboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5

TABLE T7.0-1 (Sheet 2 of 31)
PRIMARY CONTAINMENT ISOLATION DEVICES - TESTABLE PENETRATIONS

PEN. NO.	DESCRIPTION	BARRIER ORIENT.	BARRIER TYPE (MPL)	VALVE TYPE (A)	TEST TYPE (B)	POWER TO OPEN (A), (C)	POWER TO CLOSE (A), (C)	ISOL. GP. (D)	MAX. OPER. TIME (sec)	NORM. POS.	POS. ON ISOL. (E)	NOTES (F)
7A	Main Steam	Outboard	2B21-F028A	AO Globe	C	Air/AC/DC	Air/Spring	1	3<T<5	Open	Closed	1,4,7,27
7A	Main Steam	Outboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5
7B	Main Steam	Inboard	2B21-F022B	AO Globe	C	N2/AC/DC	N2/Spring	1	3<T<5	Open	Closed	1,4,7,27
7B	Main Steam	Inboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5
7B	Main Steam	Outboard	2B21-F028B	AO Globe	C	Air/AC/DC	Air/Spring	1	3<T<5	Open	Closed	1,4,7,27
7B	Main Steam	Outboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5
7C	Main Steam	Inboard	2B21-F022C	AO Globe	C	N2/AC/DC	N2/Spring	1	3<T<5	Open	Closed	1,4,7,27
7C	Main Steam	Inboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5
7C	Main Steam	Outboard	2B21-F028C	AO Globe	C	Air/AC/DC	Air/Spring	1	3<T<5	Open	Closed	1,4,7,27
7C	Main Steam	Outboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5
7D	Main Steam	Inboard	2B21-F022D	AO Globe	C	N2/AC/DC	N2/Spring	1	3<T<5	Open	Closed	1,4,7,27
7D	Main Steam	Inboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5
7D	Main Steam	Outboard	2B21-F028D	AO Globe	C	Air/AC/DC	Air/Spring	1	3<T<5	Open	Closed	1,4,7,27
7D	Main Steam	Outboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5
8	Condensate Drain	Inboard	2B21-F016	MO Gate	C	AC	AC	1	20	Closed	Closed	1,2,3,4,28
8	Condensate Drain	Inboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5
8	Condensate Drain	Outboard	2B21-F019	MO Gate	C	DC	DC	1	20	Closed	Closed	1,2,3,4,28
8	Condensate Drain	Outboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5
9A	Primary Feedwater	Inboard	2B21-F010A	Check	C	Process	Reverse Flow	-	-	Open	Closed	1,2,3,4,9,29,30
9A	Primary Feedwater	Inboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5
9A	Primary Feedwater	Outboard	2B21-F077A	AO Check	C	Process	Reverse Flow	-	-	Open	Closed	1,2,3,4,9,29,30
9A	Primary Feedwater	Outboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5
9B	Primary Feedwater	Inboard	2B21-F010B	Check	C	Process	Reverse Flow	-	-	Open	Closed	1,2,3,4,9,29,30
9B	Primary Feedwater	Inboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5
9B	Primary Feedwater	Outboard	2B21-F077B	AO Check	C	Process	Reverse Flow	-	-	Open	Closed	1,2,3,4,9,29,30
9B	Primary Feedwater	Outboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5
10	Steam to RCIC Turbine	Inboard	2E51-F007	MO Gate	C	AC	AC	4	20	Open	Closed	1,2,3,4
10	Steam to RCIC Turbine	Inboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5
10	Steam to RCIC Turbine	Outboard	2E51-F008	MO Gate	C	DC	DC	4	20	Open	Closed	1,2,3,4
10	Steam to RCIC Turbine	Outboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5
11	Steam to HPCI Turbine	Inboard	2E41-F002	MO Gate	C	AC	AC	3	50	Open	Closed	1,2,3,4
11	Steam to HPCI Turbine	Inboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5

TABLE T7.0-1 (Sheet 3 of 31)
PRIMARY CONTAINMENT ISOLATION DEVICES - TESTABLE PENETRATIONS

PEN. NO.	DESCRIPTION	BARRIER ORIENT.	BARRIER TYPE (MPL)	VALVE TYPE (A)	TEST TYPE (B)	POWER TO OPEN (A), (C)	POWER TO CLOSE (A), (C)	ISOL. GP. (D)	MAX. OPER. TIME (sec)	NORM. POS.	POS. ON ISOL. (E)	NOTES (F)
11	Steam to HPCI Turbine	Outboard	2E41-F003	MO Gate	C	DC	DC	3	50	Open	Closed	1,2,3,4
11	Steam to HPCI Turbine	Outboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5
12	RHR Shutdown Cooling Suction	Inboard	2E11-F008	MO Gate	C	DC	DC	6	24	Closed	Closed	1,2,3,4
12	RHR Shutdown Cooling Suction	Inboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5
12	RHR Shutdown Cooling Suction	Outboard	Closed System	-		-	-	-	-	-	-	24
12	RHR Shutdown Cooling Suction	Outboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5
13A	RHR Return to Recirc Loop	Inboard	2E11-F015A	MO Gate	C	AC/DC/Inv	AC/DC/Inv	I	63	Closed	Closed	1,2,3,4,11,19
13A	RHR Return to Recirc Loop	Inboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5
13A	RHR Return to Recirc Loop	Outboard	Closed System	-		-	-	-	-	-	-	24
13A	RHR Return to Recirc Loop	Outboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5
13B	RHR Return to Recirc Loop	Inboard	2E11-F015B	MO Gate	C	AC/DC/Inv	AC/DC/Inv	I	63	Closed	Closed	1,2,3,4,11,19
13B	RHR Return to Recirc Loop	Inboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5
13B	RHR Return to Recirc Loop	Outboard	Closed System	-		-	-	-	-	-	-	24
13B	RHR Return to Recirc Loop	Outboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5
14	RWC Supply	Inboard	2G31-F001	MO Gate	C	AC	AC	5,d	30	Open	Closed	1,2,3,4,28
14	RWC Supply	Inboard	Expansion Bellows	-	B	-	-			-	-	1,2,3,5
14	RWC Supply	Outboard	2G31-F004	MO Gate	C	DC	DC	5,d	30	Open	Closed	1,2,3,4,28
14	RWC Supply	Outboard	Expansion Bellows	-	B	-	-			-	-	1,2,3,5
15	Post LOCA H2 Recomb. Supply B	Inboard	2T49-F002B	MO Gate	C	AC	AC	-	-	Closed/KL	Closed	1,2,3,4,17
15	Post LOCA H2 Recomb. Supply B	Outboard	Closed System	-	C	-	-	-	-	-	-	31
16A	Core Spray A Return	Inboard	2E21-F005A	MO Gate	C	AC	AC	-	-	Closed	Closed	1,2,3,4,11,19
16A	Core Spray A Return	Inboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5
16A	Core Spray A Return	Outboard	Closed System	-		-	-	-	-	-	-	24
16A	Core Spray A Return	Outboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5
16B	Core Spray B Return	Inboard	2E21-F005B	MO Gate	C	AC	AC	-	-	Closed	Closed	1,2,3,4,11,19
16B	Core Spray B Return	Inboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5
16B	Core Spray B Return	Outboard	Closed System	-	-	-	-	-	-	-	-	24
16B	Core Spray B Return	Outboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5
17	RPV Head Spray	Inboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5
17	RPV Head Spray	Inboard	2E11-F023	MO Globe	C	DC	DC	-	-	Closed	Closed	1,2,3,4,26
17	RPV Head Spray	Outboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5
17	RPV Head Spray	Outboard	Closed System	-	-	-	-	-	-	-	-	24

TABLE T7.0-1 (Sheet 4 of 31)
PRIMARY CONTAINMENT ISOLATION DEVICES - TESTABLE PENETRATIONS

PEN. NO.	DESCRIPTION	BARRIER ORIENT.	BARRIER TYPE (MPL)	VALVE TYPE (A)	TEST TYPE (B)	POWER TO OPEN (A), (C)	POWER TO CLOSE (A), (C)	ISOL. GP. (D)	MAX. OPER. TIME (sec)	NORM. POS.	POS. ON ISOL. (E)	NOTES (F)
18	Clean Radwaste Pump Disch.	Inboard	2G11-F019	AO Gate	C	Air/AC	Spring	2,b	20	Closed	Closed	1,2,3,4,28
18	Clean Radwaste Pump Disch.	Outboard	2G11-F020	AO Gate	C	Air/AC	Spring	2,b	20	Closed	Closed	1,2,3,4,28
19	Dirty Radwaste Pump Disch.	Inboard	2G11-F003	AO Gate	C	Air/AC	Spring	2,b	20	Closed	Closed	1,2,3,4,28
19	Dirty Radwaste Pump Disch	Outboard	2G11-F004	AO Gate	C	Air/AC	Spring	2,b	20	Closed	Closed	1,2,3,4,28
21	Service Air	Inboard	2P51-F651	Globe	C	Hand	Hand	-	-	Closed	Closed	1,2,3,4,18
21	Service Air	Outboard	2P51-F513	Globe	C	Hand	Hand	-	-	Closed	Closed	1,2,3,4,18
22	Drywell Pneumatic Supply	Inboard	2P70-F004	SO Globe	C	Spring	AC	c	3	Open	Open	1,2,3,4,19,25
22	Drywell Pneumatic Supply	Inboard	2P70-N003	-	A	-	-	-	-	-	-	
22	Drywell Pneumatic Supply	Outboard	2P70-F005	SO Globe	C	Spring	AC	c	3	Open	Open	1,2,3,4,19,25
23	RBCCW Supply	Inboard	Closed System	-		-	-	-	-	-	-	19
23	RBCCW Supply	Outboard	2P42-F051	MO Gate	C	AC	AC	-	-	Open	Open	1,2,3,4,19
24	RBCCW Return	Inboard	Closed System	-		-	-	-	-	-	-	19
24	RBCCW Return	Outboard	2P42-F052	MO Gate	C	AC	AC	-	-	Open	Open	1,2,3,4,19
25	Drywell Purge Supply	Inboard	2T48-F307	AO Btfly	C	Air/AC	Spring	2	5	Closed	Closed	1,2,3,4
25	Drywell Purge Supply	Inboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5,39
25	Drywell N2 Makeup	Inboard	2T48-F118A	AO Globe	C	Air/AC	Spring	11	5	Open	Closed	1,2,3,4
25	Drywell Purge Supply	Outboard	2T48-F308	AO Btfly	C	Air/AC	Spring	2	5	Closed	Closed	1,2,3,4
25	Drywell Purge Supply	Outboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5,40
25	Drywell Purge Supply	Outboard	2T48-F103	AO Btfly	C	Air/AC	Spring	11	5	Closed	Closed	1,2,3,4
25	Drywell Purge Supply	Outboard	2T48-D006	Blind Flange	C	-	-	-	-	-	-	
25	Drywell N2 Makeup	Outboard	2T48-F104	AO Globe	C	Air/AC	Spring	11	5	Closed	Closed	1,2,3,4
26	Drywell Main Exhaust	Inboard	2T48-F319	AO Btfly	C	Air/AC	Spring	2	5	Closed	Closed	1,2,3,4
26	Drywell Main Exhaust	Inboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5,41
26	Drywell N2 Exhaust	Inboard	2T48-F341	AO Globe	C	Air/AC	Spring	11	10	Closed	Closed	1,2,3,4
26	Drywell Main Exhaust	Outboard	2T48-F320	AO Btfly	C	AC/Air	Spring	2	5	Closed	Closed	1,2,3,4
26	Drywell Main Exhaust	Outboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5,42
26	Drywell N2 Exhaust	Outboard	2T48-F340	AO Globe	C	Air/AC	Spring	11	10	Closed	Closed	1,2,3,4
27A	Recirc Line A Flow	Inboard	Orifice	-	-	-	-	-	-	-	-	
27A	Recirc Line A Flow	Outboard	2B31-F009B	EFCV	A	Spring	Process	-	-	Open	Open	56
27A	Recirc Line A Flow	Outboard	2B31-F009C	EFCV	A	Spring	Process	-	-	Open	Open	56
27B	Recirc Line A Flow	Inboard	Orifice	-	-	-	-	-	-	-	-	
27B	Recirc Line A Flow	Outboard	2B31-F010B	EFCV	A	Spring	Process	-	-	Open	Open	56

TABLE T7.0-1 (Sheet 5 of 31)
PRIMARY CONTAINMENT ISOLATION DEVICES - TESTABLE PENETRATIONS

PEN. NO.	DESCRIPTION	BARRIER ORIENT.	BARRIER TYPE (MPL)	VALVE TYPE (A)	TEST TYPE (B)	POWER TO OPEN (A), (C)	POWER TO CLOSE (A), (C)	ISOL. GP. (D)	MAX. OPER. TIME (sec)	NORM. POS.	POS. ON ISOL. (E)	NOTES (F)
27B	Recirc Line A Flow	Outboard	2B31-F010C	EFCV	A	Spring	Process	-	-	Open	Open	56
27C	Pump C001A Seal Purge	Inboard	2B31-F013A	Check	C	Process	Reverse Flow	-	-	Open	Closed	1,2,3,4,43
27C	Pump C001A Seal Purge	Outboard	2B31-F017A	Check	C	Process	Reverse Flow	-	-	Open	Closed	1,2,3,4,43
28	H202 Sample Return	Inboard	2P33-F004	AO Globe	C	Spring	Air/DC	10	5	Open	Closed	1,2,3,4
28	H202 Sample Return	Outboard	2P33-F012	AO Globe	C	Spring	Air/AC	10	5	Open	Closed	1,2,3,4
29A	Recirc Line B Flow	Inboard	Orifice	-	-	-	-	-	-	-	-	
29A	Recirc Line B Flow	Outboard	2B31-F011A	EFCV	A	Spring	Process	-	-	Open	Open	56
29A	Recirc Line B Flow	Outboard	2B31-F011D	EFCV	A	Spring	Process	-	-	Open	Open	56
29B	Recirc Line B Flow	Inboard	Orifice	-	-	-	-	-	-	-	-	
29B	Recirc Line B Flow	Outboard	2B31-F012A	EFCV	A	Spring	Process	-	-	Open	Open	56
29B	Recirc Line B Flow	Outboard	2B31-F012D	EFCV	A	Spring	Process	-	-	Open	Open	56
30A	Recirc Line B Press (spared)	Inboard	2B31-F058B	Globe	C	Hand	Hand	-	-	Closed	Closed	1,2,3,4
30A	Recirc Line B Press (spared)	Outboard	Swagelock Cap	-	C	-	-	-	-	-	-	32
30B	Recirc Pump B Disch Press	Inboard	Orifice	-	-	-	-	-	-	-	-	
30B	Recirc Pump B Disch Press	Outboard	2B31-F040B	EFCV	A	Spring	Process	-	-	Open	Open	56
30C	Recirc Pump B Suct Press	Inboard	Orifice	-	-	-	-	-	-	-	-	
30C	Recirc Pump B Suct Press	Outboard	2B31-F040D	EFCV	A	Spring	Process	-	-	Open	Open	56
30D	Recirc Pump B Seal 2	Inboard	Orifice	-	-	-	-	-	-	-	-	
30D	Recirc Pump B Seal 2	Outboard	2B31-F003B	EFCV	A	Spring	Process	-	-	Open	Open	56
30E	Recirc Pump B Seal 1	Inboard	Orifice	-	-	-	-	-	-	-	-	
30E	Recirc Pump B Seal 1	Outboard	2B31-F004B	EFCV	A	Spring	Process	-	-	Open	Open	56
31A	Recirc Line A (spared)	Inboard	2B31-F058A	Globe	C	Hand	Hand	-	-	Closed	Closed	1,2,3,4
31A	Recirc Line A (spared)	Outboard	Swagelock Cap	-	C	-	-	-	-	-	-	32
31B	Recirc Pump A Disch Press	Inboard	Orifice	-	-	-	-	-	-	-	-	
31B	Recirc Pump A Disch Press	Outboard	2B31-F040A	EFCV	A	Spring	Process	-	-	Open	Open	56
31C	Recirc Pump A Suct Press	Inboard	Orifice	-	-	-	-	-	-	-	-	
31C	Recirc Pump A Suct Press	Outboard	2B31-F040C	EFCV	A	Spring	Process	-	-	Open	Open	56
31D	Recirc Pump A Seal 2	Inboard	Orifice	-	-	-	-	-	-	-	-	
31D	Recirc Pump A Seal 2	Outboard	2B31-F003A	EFCV	A	Spring	Process	-	-	Open	Open	56
31E	Recirc Pump A Seal 1	Inboard	Orifice	-	-	-	-	-	-	-	-	
31E	Recirc Pump A Seal 1	Outboard	2B31-F004A	EFCV	A	Spring	Process	-	-	Open	Open	56
32A	Drywell Pressure	Inboard	Orifice	-	-	-	-	-	-	-	-	

TABLE T7.0-1 (Sheet 6 of 31)
PRIMARY CONTAINMENT ISOLATION DEVICES - TESTABLE PENETRATIONS

PEN. NO.	DESCRIPTION	BARRIER ORIENT.	BARRIER TYPE (MPL)	VALVE TYPE (A)	TEST TYPE (B)	POWER TO OPEN (A), (C)	POWER TO CLOSE (A), (C)	ISOL. GP. (D)	MAX. OPER. TIME (sec)	NORM. POS.	POS. ON ISOL. (E)	NOTES (F)
32A	Drywell Pressure	Outboard	2E11-F041D	AO Globe	C	Spring	Air/AC	-	-	Open	Open	1,2,3,4,33
32C	Drywell Pressure	Inboard	Orifice	-	-	-	-	-	-	-	-	
32C	Drywell Pressure	Outboard	2E11-F041B	AO Globe	C	Spring	Air/AC	-	-	Open	Open	1,2,3,4,33
33A	Main Steam Line A Flow	Inboard	Orifice	-	-	-	-	-	-	-	-	
33A	Main Steam Line A Flow	Outboard	2B21-F072A	EFCV	A	Spring	Process	-	-	Open	Open	56
33B	Main Steam Line B Flow	Inboard	Orifice	-	-	-	-	-	-	-	-	
33B	Main Steam Line B Flow	Outboard	2B21-F072B	EFCV	A	Spring	Process	-	-	Open	Open	56
33C	Main Steam Line B Flow	Inboard	Orifice	-	-	-	-	-	-	-	-	
33C	Main Steam Line B Flow	Outboard	2B21-F071B	EFCV	A	Spring	Process	-	-	Open	Open	56
33D	Main Steam Line A Flow	Inboard	Orifice	-	-	-	-	-	-	-	-	
33D	Main Steam Line A Flow	Outboard	2B21-F071A	EFCV	A	Spring	Process	-	-	Open	Open	56
33E	Main Steam Line C Flow	Inboard	Orifice	-	-	-	-	-	-	-	-	
33E	Main Steam Line C Flow	Outboard	2B21-F071C	EFCV	A	Spring	Process	-	-	Open	Open	56
33F	Main Steam Line C Flow	Inboard	Orifice	-	-	-	-	-	-	-	-	
33F	Main Steam Line C Flow	Outboard	2B21-F072C	EFCV	A	Spring	Process	-	-	Open	Open	56
34A	Main Steam Line D Flow	Inboard	Orifice	-	-	-	-	-	-	-	-	
34A	Main Steam Line D Flow	Outboard	2B21-F072D	EFCV	A	Spring	Process	-	-	Open	Open	56
34B	Main Steam Line D Flow	Inboard	Orifice	-	-	-	-	-	-	-	-	
34B	Main Steam Line D Flow	Outboard	2B21-F071D	EFCV	A	Spring	Process	-	-	Open	Open	56
34C	ILRT Verification Flow	Inboard	2T23-F004	Globe	C	Hand	Hand	-	-	Closed	Closed	1,2,3,4,18
34C	ILRT Verification Flow	Outboard	2T23-F005	Globe	C	Hand	Hand	-	-	Closed	Closed	1,2,3,4,18
34D	Drywell Pressure	Inboard	Orifice	-	-	-	-	-	-	-	-	
34D	Drywell Pressure	Outboard	2T48-F363A	AO Globe	C	Spring	Air/AC	-	-	Open	Open	1,2,3,4,33,56
34E	RCIC Steam Line DP	Inboard	Orifice	-	-	-	-	-	-	-	-	
34E	RCIC Steam Line DP	Outboard	2E51-F044C	EFCV	A	Spring	Process	-	-	Open	Open	56
34F	RCIC Steam Line DP	Inboard	Orifice	-	-	-	-	-	-	-	-	
34F	RCIC Steam Line DP	Outboard	2E51-F044A	EFCV	A	Spring	Process	-	-	Open	Open	56
35A	TIP Drive D	Inboard	Ball Vlv For J004D	Ball	C	AC	AC	e	NA	Closed	Closed	1,2,3,4,13
35A	TIP Drive D	Inboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5,54
35A	TIP Drive D	Outboard	Shear Vlv For J004D	Shear	-	-	DC,Explosive	-	-	Open	As	13,34
35B	TIP Drive A	Inboard	Ball Vlv For J004A	Ball	C	AC	AC	e	NA	Closed	Closed	1,2,3,4,13
35B	TIP Drive A	Inboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5,54

TABLE T7.0-1 (Sheet 7 of 31)
PRIMARY CONTAINMENT ISOLATION DEVICES - TESTABLE PENETRATIONS

PEN. NO.	DESCRIPTION	BARRIER ORIENT.	BARRIER TYPE (MPL)	VALVE TYPE (A)	TEST TYPE (B)	POWER TO OPEN (A), (C)	POWER TO CLOSE (A), (C)	ISOL. GP. (D)	MAX. OPER. TIME (sec)	NORM. POS.	POS. ON ISOL. (E)	NOTES (F)
35B	TIP Drive A	Outboard	Shear Vlv For J004A	Shear	-	-	DC,Explosive	-	-	Open	As	13,34
35C	TIP Drive C	Inboard	Ball Vlv For J004C	Ball	C	AC	AC	e	NA	Closed	Closed	1,2,3,4,13
35C	TIP Drive C	Inboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5,54
35C	TIP Drive C	Outboard	Shear Vlv For J004C	Shear	-	-	DC,Explosive	-	-	Open	As	13,34
35D	TIP Drive B	Inboard	Ball Vlv For J004B	Ball	C	AC	AC	e	NA	Closed	Closed	1,2,3,4,13
35D	TIP Drive B	Inboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5,54
35D	TIP Drive B	Outboard	Shear Vlv For J004B	Shear	-	-	DC,Explosive	-	-	Open	As	13,34
35E	TIP N2 Purge	Inboard	2C51-F3017	Check	C	Process	Reverse Flow	-	-	Open	Closed	1,2,3,4
35E	TIP N2 Purge	Inboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5,54
35E	TIP N2 Purge	Outboard	2C51-F3012	Solenoid	C	DC	-	2,b	NA	Open	Closed	1,2,3,4
35E	TIP N2 Purge	Outboard	2C51-R751	-	A	-	-	-	-	-	-	
35E	TIP N2 Purge	Outboard	2C51-R752	-	A	-	-	-	-	-	-	
37A	CRD Insert (typical 38)	Inboard	-	-		-	-	-	-	-	-	
37A	CRD Spare (typical 40)	Inboard	Welded Cap	-	A	-	-	-	-	-	-	
37A	CRD Insert (typical 38)	Outboard	2C11-D001-120	Solenoid	A	AC	Spring	-	-	Closed	Closed	10,35
37A	CRD Insert (typical 38)	Outboard	2C11-D001-123	Solenoid	A	AC	Spring	-	-	Closed	Closed	10,35
37A	CRD Insert (typical 38)	Outboard	2C11-D001-126	AO Globe	A	Spring	Air/AC	-	-	Closed	Open	10,35
37A	CRD Insert (typical 38)	Outboard	2C11-D001-138	Check	A	Process	Process	-	-	Open	Closed	10,35
37A	CRD Spare (typical 40)	Outboard	-	-	-	-	-	-	-	-	-	
37B	CRD Insert (typical 31)	Inboard	-	-		-	-	-	-	-	-	
37B	CRD Spare (typical 33)	Inboard	Welded Cap	-	A	-	-	-	-	-	-	
37B	CRD Insert (typical 31)	Outboard	2C11-D001-120	Solenoid	A	AC	Spring	-	-	Closed	Closed	10,35
37B	CRD Insert (typical 31)	Outboard	2C11-D001-123	Solenoid	A	AC	Spring	-	-	Closed	Closed	10,35
37B	CRD Insert (typical 31)	Outboard	2C11-D001-126	AO Globe	A	Spring	Air/AC	-	-	Closed	Open	10,35
37B	CRD Insert (typical 31)	Outboard	2C11-D001-138	Check	A	Process	Process	-	-	Open	Closed	10,35
37B	CRD Spare (typical 33)	Outboard	-	-	-	-	-	-	-	-	-	
37C	CRD Insert (typical 31)	Inboard	-	-		-	-	-	-	-	-	
37C	CRD Spare (typical 33)	Inboard	Welded Cap	-	A	-	-	-	-	-	-	
37C	CRD Insert (typical 31)	Outboard	2C11-D001-120	Solenoid	A	AC	Spring	-	-	Closed	Closed	10,35
37C	CRD Insert (typical 31)	Outboard	2C11-D001-123	Solenoid	A	AC	Spring	-	-	Closed	Closed	10,35
37C	CRD Insert (typical 31)	Outboard	2C11-D001-126	AO Globe	A	Spring	Air/AC	-	-	Closed	Open	10,35
37C	CRD Insert (typical 31)	Outboard	2C11-D001-138	Check	A	Process	Process	-	-	Open	Closed	10,35

TABLE T7.0-1 (Sheet 8 of 31)
PRIMARY CONTAINMENT ISOLATION DEVICES - TESTABLE PENETRATIONS

PEN. NO.	DESCRIPTION	BARRIER ORIENT.	BARRIER TYPE (MPL)	VALVE TYPE (A)	TEST TYPE (B)	POWER TO OPEN (A), (C)	POWER TO CLOSE (A), (C)	ISOL. GP. (D)	MAX. OPER. TIME (sec)	NORM. POS.	POS. ON ISOL. (E)	NOTES (F)
37C	CRD Spare (typical 33)	Outboard	-	-	-	-	-	-	-	-	-	
37D	CRD Insert (typical 37)	Inboard	-	-		-	-	-	-	-	-	
37D	CRD Spare (typical 39)	Inboard	Welded Cap	-	A	-	-	-	-	-	-	
37D	CRD Insert (typical 37)	Outboard	2C11-D001-120	Solenoid	A	AC	Spring	-	-	Closed	Closed	10,35
37D	CRD Insert (typical 37)	Outboard	2C11-D001-123	Solenoid	A	AC	Spring	-	-	Closed	Closed	10,35
37D	CRD Insert (typical 37)	Outboard	2C11-D001-126	AO Globe	A	Spring	Air/AC	-	-	Closed	Open	10,35
37D	CRD Insert (typical 37)	Outboard	2C11-D001-138	Check	A	Process	Process	-	-	Open	Closed	10,35
37D	CRD Spare (typical 39)	Outboard	-	-	-	-	-	-	-	-	-	
38A	CRD Insert (typical 38)	Inboard	-	-		-	-	-	-	-	-	
38A	CRD Spare (typical 40)	Inboard	Welded Cap	-	A	-	-	-	-	-	-	
38A	CRD Insert (typical 38)	Outboard	2C11-D001-121	Solenoid	A	AC	Spring	-	-	Closed	Closed	10,35
38A	CRD Insert (typical 38)	Outboard	2C11-D001-122	Solenoid	A	AC	Spring	-	-	Closed	Closed	10,35
38A	CRD Insert (typical 38)	Outboard	2C11-D001-127	AO Globe	A	Spring	Air/AC	-	-	Closed	Open	10,35
38A	CRD Spare (typical 40)	Outboard	-	-	-	-	-	-	-	-	-	
38B	CRD Insert (typical 31)	Inboard	-	-		-	-	-	-	-	-	
38B	CRD Spare (typical 33)	Inboard	Welded Cap	-	A	-	-	-	-	-	-	
38B	CRD Insert (typical 31)	Outboard	2C11-D001-121	Solenoid	A	AC	Spring	-	-	Closed	Closed	10,35
38B	CRD Insert (typical 31)	Outboard	2C11-D001-122	Solenoid	A	AC	Spring	-	-	Closed	Closed	10,35
38B	CRD Insert (typical 31)	Outboard	2C11-D001-127	AO Globe	A	Spring	Air/AC	-	-	Closed	Open	10,35
38B	CRD Spare (typical 33)	Outboard	-	-	-	-	-	-	-	-	-	
38C	CRD Insert (typical 31)	Inboard	-	-		-	-	-	-	-	-	
38C	CRD Spare (typical 33)	Inboard	Welded Cap	-	A	-	-	-	-	-	-	
38C	CRD Insert (typical 31)	Outboard	2C11-D001-121	Solenoid	A	AC	Spring	-	-	Closed	Closed	10,35
38C	CRD Insert (typical 31)	Outboard	2C11-D001-122	Solenoid	A	AC	Spring	-	-	Closed	Closed	10,35
38C	CRD Insert (typical 31)	Outboard	2C11-D001-127	AO Globe	A	Spring	Air/AC	-	-	Closed	Open	10,35
38C	CRD Spare (typical 33)	Outboard	-	-	-	-	-	-	-	-	-	
38D	CRD Insert (typical 37)	Inboard	-	-		-	-	-	-	-	-	
38D	CRD Spare (typical 39)	Inboard	Welded Cap	-	A	-	-	-	-	-	-	
38D	CRD Insert (typical 37)	Outboard	2C11-D001-121	Solenoid	A	AC	Spring	-	-	Closed	Closed	10,35
38D	CRD Insert (typical 37)	Outboard	2C11-D001-122	Solenoid	A	AC	Spring	-	-	Closed	Closed	10,35
38D	CRD Insert (typical 37)	Outboard	2C11-D001-127	AO Globe	A	Spring	Air/AC	-	-	Closed	Open	10,35
38D	CRD Spare (typical 39)	Outboard	-	-	-	-	-	-	-	-	-	

TABLE T7.0-1 (Sheet 9 of 31)
PRIMARY CONTAINMENT ISOLATION DEVICES - TESTABLE PENETRATIONS

PEN. NO.	DESCRIPTION	BARRIER ORIENT.	BARRIER TYPE (MPL)	VALVE TYPE (A)	TEST TYPE (B)	POWER TO OPEN (A), (C)	POWER TO CLOSE (A), (C)	ISOL. GP. (D)	MAX. OPER. TIME (sec)	NORM. POS.	POS. ON ISOL. (E)	NOTES (F)
39A	Containment Spray	Inboard	2E11-F016A	MO Globe	C	AC	AO	g	10	Closed/KL	Closed	1,2,3,4,8
39A	Containment Spray	Outboard	Closed System	-	-	-	-	-	-	-	-	24
39B	Containment Spray	Inboard	2E11-F016B	MO Globe	C	AC	AC	g	10	Closed/KL	Closed	1,2,3,4,8
39B	Containment Spray	Outboard	Closed System	-	-	-	-	-	-	-	-	24
40A (C)	Press Above Core Plate	Inboard	Orifice	-	-	-	-	-	-	-	-	
40A (C)	Press Above Core Plate	Outboard	2E21-F018C	EFCV	A	Spring	Process	-	-	Open	Open	56
40A (D)	Press Below Core Plate	Inboard	Orifice	-	-	-	-	-	-	-	-	
40A (D)	Press Below Core Plate	Outboard	2B21-F061	EFCV	A	Spring	Process	-	-	Open	Open	56
40B (A)	Jet Pump Inst	Inboard	Orifice	-	-	-	-	-	-	-	-	
40B (A)	Jet Pump Inst	Outboard	2B21-F053A	EFCV	A	Spring	Process	-	-	Open	Open	56
40B (B)	Jet Pump Inst	Inboard	Orifice	-	-	-	-	-	-	-	-	
40B (B)	Jet Pump Inst	Outboard	2B21-F059C	EFCV	A	Spring	Process	-	-	Open	Open	56
40B (C)	Jet Pump Inst	Inboard	Orifice	-	-	-	-	-	-	-	-	
40B (C)	Jet Pump Inst	Outboard	2B21-F051A	EFCV	A	Spring	Process	-	-	Open	Open	56
40B (D)	Jet Pump Inst	Inboard	Orifice	-	-	-	-	-	-	-	-	
40B (D)	Jet Pump Inst	Outboard	2B21-F059E	EFCV	A	Spring	Process	-	-	Open	Open	56
40B (E)	Jet Pump Inst	Inboard	Orifice	-	-	-	-	-	-	-	-	
40B (E)	Jet Pump Inst	Outboard	2B21-F059G	EFCV	A	Spring	Process	-	-	Open	Open	56
40B (F)	Jet Pump Inst	Inboard	Orifice	-	-	-	-	-	-	-	-	
40B (F)	Jet Pump Inst	Outboard	2B21-F059A	EFCV	A	Spring	Process	-	-	Open	Open	56
40C (C)	Press Above Core Plate	Inboard	Orifice	-	-	-	-	-	-	-	-	
40C (C)	Press Above Core Plate	Outboard	2B21-F055	EFCV	A	Spring	Process	-	-	Open	Open	56
40C (D)	Press Below Core Plate	Inboard	Orifice	-	-	-	-	-	-	-	-	
40C (D)	Press Below Core Plate	Outboard	2B21-F057	EFCV	A	Spring	Process	-	-	Open	Open	56
40D (A)	HPCI Steam Line DP	Inboard	Orifice	-	-	-	-	-	-	-	-	
40D (A)	HPCI Steam Line DP	Outboard	2E41-F024B	EFCV	A	Spring	Process	-	-	Open	Open	56
40D (B)	HPCI Steam Line DP	Inboard	Orifice	-	-	-	-	-	-	-	-	
40D (B)	HPCI Steam Line DP	Outboard	2E41-F024D	EFCV	A	Spring	Process	-	-	Open	Open	56
40D (F)	CS Diff Press Inst	Inboard	Orifice	-	-	-	-	-	-	-	-	
40D (F)	CS Diff Press Inst	Outboard	2E21-F018B	EFCV	A	Spring	Process	-	-	Open	Open	56
41	Reactor Water Sample Line	Inboard	2B31-F019	AO Globe	C	N2/AC	Spring	1,a	5	Open	Closed	1,2,3,4
41	Reactor Water Sample Line	Outboard	2B31-F020	AO Globe	C	Air/AC	Spring	1,a	5	Open	Closed	1,2,3,4

TABLE T7.0-1 (Sheet 10 of 31)
PRIMARY CONTAINMENT ISOLATION DEVICES - TESTABLE PENETRATIONS

PEN. NO.	DESCRIPTION	BARRIER ORIENT.	BARRIER TYPE (MPL)	VALVE TYPE (A)	TEST TYPE (B)	POWER TO OPEN (A), (C)	POWER TO CLOSE (A), (C)	ISOL. GP. (D)	MAX. OPER. TIME (sec)	NORM. POS.	POS. ON ISOL. (E)	NOTES (F)
42	Standby Liquid Control	Inboard	2C41-F007	Check	C	Process	Reverse Flow	-	-	Closed	Closed	1,2,3,4,44
42	Standby Liquid Control	Inboard	Expansion Bellows	-	-	-	-	-	-	-	-	1,2,3,5
42	Standby Liquid Control	Outboard	2C41-F006	Check	C	Process	Reverse Flow	-	-	Closed	Closed	1,2,3,4,44
42	Standby Liquid Control	Outboard	Expansion Bellows	-	-	-	-	-	-	-	-	1,2,3,5
43	Drywell Test and Fill	Inboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5,54
43	Drywell Test and Fill	Outboard	-	-	-	-	-	-	-	-	-	
44	Drywell N2 Makeup Inlet	Inboard	2T48-F322	AO Globe	C	N2/AC	Spring	-	-	Closed	Closed	1,2,3,4,20
44	Drywell N2 Makeup Inlet	Outboard	2T48-F321	AO Globe	C	N2/AC	Spring	-	-	Closed	Closed	1,2,3,4,17,20
45A	Jet Pump Inst	Inboard	Orifice	-	-	-	-	-	-	-	-	
45A	Jet Pump Inst	Outboard	2B21-F053C	EFCV	A	Spring	Process	-	-	Open	Open	56
45B	Jet Pump Inst	Inboard	Orifice	-	-	-	-	-	-	-	-	
45B	Jet Pump Inst	Outboard	2B21-F059L	EFCV	A	Spring	Process	-	-	Open	Open	56
45C	Jet Pump Inst	Inboard	Orifice	-	-	-	-	-	-	-	-	
45C	Jet Pump Inst	Outboard	2B21-F059R	EFCV	A	Spring	Process	-	-	Open	Open	56
45D	Jet Pump Inst	Inboard	Orifice	-	-	-	-	-	-	-	-	
45D	PASS	Inboard	2B21-F111	AO Gate	C	Air/AC	Spring	-	-	Closed	Closed	1,2,3,4,45
45D	Jet Pump Inst	Outboard	2B21-F051C	EFCV	A	Spring	Process	-	-	Open	Open	56
45D	PASS	Outboard	2B21-F112	AO Gate	C	Air/AC	Spring	-	-	Closed	Closed	1,2,3,4,45
45E	Jet Pump Inst	Inboard	Orifice	-	-	-	-	-	-	-	-	
45E	Jet Pump Inst	Outboard	2B21-F059T	EFCV	A	Spring	Process	-	-	Open	Open	56
45F	Jet Pump Inst	Inboard	Orifice	-	-	-	-	-	-	-	-	
45F	Jet Pump Inst	Outboard	2B21-F059N	EFCV	A	Spring	Process	-	-	Open	Open	56
46	Demineralized Water	Inboard	2P21-F034	Gate	C	Hand	Hand	-	-	Closed/LC	Closed	1,2,3,4,18
46	Demineralized Water	Outboard	2P21-F032	Gate	C	Hand	Hand	-	-	Closed/LC	Closed	1,2,3,4,18
47	Chilled Water Supply	Inboard	Closed System	-	-	-	-	-	-	-	-	19,36
47	Chilled Water Supply	Outboard	2P64-F045	MO Globe	C	AC	AC	-	-	Open	Open	1,2,3,4,19
48	Chilled Water Return	Inboard	Closed System	-	-	-	-	-	-	-	-	19,36
48	Chilled Water Return	Outboard	2P64-F047	MO Globe	C	AC	AC	-	-	Open	Open	1,2,3,4,19
49A	Jet Pump Inst	Inboard	Orifice	-	-	-	-	-	-	-	-	
49A	Jet Pump Inst	Outboard	2B21-F053B	EFCV	A	Spring	Process	-	-	Open	Open	56
49B	Jet Pump Inst	Inboard	Orifice	-	-	-	-	-	-	-	-	
49B	Jet Pump Inst	Outboard	2B21-F059D	EFCV	A	Spring	Process	-	-	Open	Open	56

TABLE T7.0-1 (Sheet 11 of 31)
PRIMARY CONTAINMENT ISOLATION DEVICES - TESTABLE PENETRATIONS

PEN. NO.	DESCRIPTION	BARRIER ORIENT.	BARRIER TYPE (MPL)	VALVE TYPE (A)	TEST TYPE (B)	POWER TO OPEN (A), (C)	POWER TO CLOSE (A), (C)	ISOL. GP. (D)	MAX. OPER. TIME (sec)	NORM. POS.	POS. ON ISOL. (E)	NOTES (F)
49C	Jet Pump Inst	Inboard	Orifice	-	-	-	-	-	-	-	-	
49C	Jet Pump Inst	Outboard	2B21-F051B	EFCV	A	Spring	Process	-	-	Open	Open	56
49D	Jet Pump Inst	Inboard	Orifice	-	-	-	-	-	-	-	-	
49D	Jet Pump Inst	Outboard	2B21-F059F	EFCV	A	Spring	Process	-	-	Open	Open	56
49E	Jet Pump Inst	Inboard	Orifice	-	-	-	-	-	-	-	-	
49E	Jet Pump Inst	Outboard	2B21-F059H	EFCV	A	Spring	Process	-	-	Open	Open	56
49F	Jet Pump Inst	Inboard	Orifice	-	-	-	-	-	-	-	-	
49F	Jet Pump Inst	Outboard	2B21-F059B	EFCV	A	Spring	Process	-	-	Open	Open	56
50A	Jet Pump Inst	Inboard	Orifice	-	-	-	-	-	-	-	-	
50A	Jet Pump Inst	Outboard	2B21-F053D	EFCV	A	Spring	Process	-	-	Open	Open	56
50B	Jet Pump Inst	Inboard	Orifice	-	-	-	-	-	-	-	-	
50B	Jet Pump Inst	Outboard	2B21-F059M	EFCV	A	Spring	Process	-	-	Open	Open	56
50C	Jet Pump Inst	Inboard	Orifice	-	-	-	-	-	-	-	-	
50C	Jet Pump Inst	Outboard	2B21-F059S	EFCV	A	Spring	Process	-	-	Open	Open	56
50D	Jet Pump Inst	Inboard	Orifice	-	-	-	-	-	-	-	-	
50D	Jet Pump Inst	Outboard	2B21-F051D	EFCV	A	Spring	Process	-	-	Open	Open	56
50E	Jet Pump Inst	Inboard	Orifice	-	-	-	-	-	-	-	-	
50E	Jet Pump Inst	Outboard	2B21-F059U	EFCV	A	Spring	Process	-	-	Open	Open	56
50F	Jet Pump Inst	Inboard	Orifice	-	-	-	-	-	-	-	-	
50F	Jet Pump Inst	Outboard	2B21-F059P	EFCV	A	Spring	Process	-	-	Open	Open	56
51A	RCIC Steam Line DP	Inboard	Orifice	-	-	-	-	-	-	-	-	
51A	RCIC Steam Line DP	Outboard	2E51-F044D	EFCV	A	Spring	Process	-	-	Open	Open	56
51B	RCIC Steam Line DP	Inboard	Orifice	-	-	-	-	-	-	-	-	
51B	RCIC Steam Line DP	Outboard	2E51-F044B	EFCV	A	Spring	Process	-	-	Open	Open	56
51C	Drywell Pneumatic Supply	Inboard	2P70-F066	SO Globe	C	Spring	AC	C	3	Open	Open	1,2,3,4,19,25
51C	Drywell Pneumatic Supply	Inboard	2P70-N016	-	A	-	-	-	-	-	-	
51C	Drywell Pneumatic Supply	Outboard	2P70-F067	SO Globe	C	Spring	AC	C	3	Open	Open	1,2,3,4,19,25
51D	Drywell Pressure	Inboard	Orifice	-	-	-	-	-	-	-	-	
51D	Drywell Pressure	Outboard	2T48-F363B	AO Globe	C	Spring	Air/AC	-	-	Open	Open	1,2,3,4,33,56
51E	Main Steam Line D Flow	Inboard	Orifice	-	-	-	-	-	-	-	-	
51E	Main Steam Line D Flow	Outboard	2B21-F070D	EFCV	A	Spring	Process	-	-	Open	Open	56
51F	Main Steam Line D Flow	Inboard	Orifice	-	-	-	-	-	-	-	-	

TABLE T7.0-1 (Sheet 12 of 31)
PRIMARY CONTAINMENT ISOLATION DEVICES - TESTABLE PENETRATIONS

PEN. NO.	DESCRIPTION	BARRIER ORIENT.	BARRIER TYPE (MPL)	VALVE TYPE (A)	TEST TYPE (B)	POWER TO OPEN (A), (C)	POWER TO CLOSE (A), (C)	ISOL. GP. (D)	MAX. OPER. TIME (sec)	NORM. POS.	POS. ON ISOL. (E)	NOTES (F)
51F	Main Steam Line D Flow	Outboard	2B21-F073D	EFCV	A	Spring	Process	-	-	Open	Open	56
52A	Main Steam Line A Flow	Inboard	Orifice	-	-	-	-	-	-	-	-	
52A	Main Steam Line A Flow	Outboard	2B21-F073A	EFCV	A	Spring	Process	-	-	Open	Open	56
52B	Main Steam Line C Flow	Inboard	Orifice	-	-	-	-	-	-	-	-	
52B	Main Steam Line C Flow	Outboard	2B21-F073C	EFCV	A	Spring	Process	-	-	Open	Open	56
52C	Main Steam Line C Flow	Inboard	Orifice	-	-	-	-	-	-	-	-	
52C	Main Steam Line C Flow	Outboard	2B21-F070C	EFCV	A	Spring	Process	-	-	Open	Open	56
52D	Main Steam Line A Flow	Inboard	Orifice	-	-	-	-	-	-	-	-	
52D	Main Steam Line A Flow	Outboard	2B21-F070A	EFCV	A	Spring	Process	-	-	Open	Open	56
52E	Main Steam Line B Flow	Inboard	Orifice	-	-	-	-	-	-	-	-	
52E	Main Steam Line B Flow	Outboard	2B21-F070B	EFCV	A	Spring	Process	-	-	Open	Open	56
52F	Main Steam Line B Flow	Inboard	Orifice	-	-	-	-	-	-	-	-	
52F	Main Steam Line B Flow	Outboard	2B21-F073B	EFCV	A	Spring	Process	-	-	Open	Open	56
54A	Drywell Pressure	Inboard	Orifice	-	-	-	-	-	-	-	-	
54A	Drywell Pressure	Outboard	2E11-F041C	AO Globe	C	Spring	Air/AC	-	-	Open	Open	1,2,3,4,33
54C	Drywell Pressure	Inboard	Orifice	-	-	-	-	-	-	-	-	
54C	Drywell Pressure	Outboard	2E11-F041A	AO Globe	C	Spring	Air/AC	-	-	Open	Open	1,2,3,4,33
55	Chemical Pump Discharge	Inboard	2G11-F852	Gate	C	Hand	Hand	-	-	Closed/LC	Closed	1,2,3,4,18,28
55	Chemical Pump Discharge	Outboard	2G11-F853	Gate	C	Hand	Hand	-	-	Closed/LC	Closed	1,2,3,4,18,28
56B	RPV Level and DP Inst	Inboard	Orifice	-	-	-	-	-	-	-	-	
56B	RPV Level and DP Inst	Outboard	2B21-F047A	EFCV	A	Spring	Process	-	-	Open	Open	56
56C	RPV Level Inst	Inboard	Orifice	-	-	-	-	-	-	-	-	
56C	RPV Level Inst	Outboard	2B21-F045A	EFCV	A	Spring	Process	-	-	Open	Open	56
56D	RPV Level and Pressure	Inboard	Orifice	-	-	-	-	-	-	-	-	
56D	RPV Level and Pressure	Outboard	2B21-F049A	EFCV	A	Spring	Process	-	-	Open	Open	56
56E	RPV Level and Pressure	Inboard	Orifice	-	-	-	-	-	-	-	-	
56E	RPV Level and Pressure	Outboard	2B21-F043A	EFCV	A	Spring	Process	-	-	Open	Open	56
56F	RPV Level Inst	Inboard	Orifice	-	-	-	-	-	-	-	-	
56F	RPV Level Inst	Outboard	2B21-F041	EFCV	A	Spring	Process	-	-	Open	Open	56
57A	Recirc Line B Flow	Inboard	Orifice	-	-	-	-	-	-	-	-	
57A	Recirc Line B Flow	Outboard	2B31-F011B	EFCV	A	Spring	Process	-	-	Open	Open	56
57A	Recirc Line B Flow	Outboard	2B31-F011C	EFCV	A	Spring	Process	-	-	Open	Open	56

TABLE T7.0-1 (Sheet 13 of 31)
PRIMARY CONTAINMENT ISOLATION DEVICES - TESTABLE PENETRATIONS

PEN. NO.	DESCRIPTION	BARRIER ORIENT.	BARRIER TYPE (MPL)	VALVE TYPE (A)	TEST TYPE (B)	POWER TO OPEN (A), (C)	POWER TO CLOSE (A), (C)	ISOL. GP. (D)	MAX. OPER. TIME (sec)	NORM. POS.	POS. ON ISOL. (E)	NOTES (F)
57B	Recirc Line B Flow	Inboard	Orifice	-		-	-	-	-	-	-	
57B	Recirc Line B Flow	Outboard	2B31-F012B	EFCV	A	Spring	Process	-	-	Open	Open	56
57B	Recirc Line B Flow	Outboard	2B31-F012C	EFCV	A	Spring	Process	-	-	Open	Open	56
57C	Recirc Pump B Seal Purge	Inboard	2B31-F013B	Check	C	Process	Reverse Flow	-	-	Open	Closed	1,2,3,4,43
57C	Recirc Pump B Seal Purge	Outboard	2B31-F017B	Check	C	Process	Reverse Flow	-	-	Open	Closed	1,2,3,4,43
59B	RPV Level and DP Inst	Inboard	Orifice	-	-	-	-	-	-	-	-	
59B	RPV Level and DP Inst	Outboard	2B21-F047B	EFCV	A	Spring	Process	-	-	Open	Open	56
59C	RPV Level Inst	Inboard	Orifice	-	-	-	-	-	-	-	-	
59C	RPV Level Inst	Outboard	2B21-F045B	EFCV	A	Spring	Process	-	-	Open	Open	56
59D	RPV Level and Pressure	Inboard	Orifice	-	-	-	-	-	-	-	-	
59D	RPV Level and Pressure	Outboard	2B21-F049B	EFCV	A	Spring	Process	-	-	Open	Open	56
59E	RPV Level and Pressure	Inboard	Orifice	-	-	-	-	-	-	-	-	
59E	RPV Level and Pressure	Outboard	2B21-F043B	EFCV	A	Spring	Process	-	-	Open	Open	56
60A	H2O2 Sample Supply	Inboard	2P33-F003	AO Globe	C	Spring	DC/Air	10	5	Closed	Closed	1,2,3,4
60A	H2O2 Sample Supply	Outboard	2P33-F011	AO Globe	C	Spring	Air/AC	10	5	Closed	Closed	1,2,3,4
60B	FPM Sample Return	Inboard	2D11-F050	AO Globe	C	Air/DC	Spring	11	5	Open	Closed	1,2,3,4
60B	FPM Sample Return	Outboard	2D11-F052	AO Globe	C	Air/AC	Spring	11	5	Open	Closed	1,2,3,4
61A	Post LOCA H2 Recomb Supply A	Inboard	2T49-F002A	MO Gate	C	AC	AC	-	-	Closed/KL	Closed	1,2,3,4,17
61A	Post LOCA H2 Recomb Supply A	Outboard	Closed System	-	C	-	-	-	-	-	-	31
62	FPM Sample Supply	Inboard	2D11-F051	AO Globe	C	Air/AC	Spring	11	5	Open	Closed	1,2,3,4
62	FPM Sample Supply	Outboard	2D11-F053	AO Globe	C	Air/AC	Spring	11	5	Open	Closed	1,2,3,4
63	Drywell Pneumatic Suction	Inboard	2P70-F002	AO Globe	C	Air/AC	Spring	11	5	Closed	Closed	1,2,3,4
63	Drywell Pneumatic Suction	Outboard	2P70-F003	AO Globe	C	Air/AC	Spring	11	5	Closed	Closed	1,2,3,4
64	H2O2 Sample Return	Inboard	2P33-F005	AO Globe	C	Spring	DC/Air	10	5	Open	Closed	1,2,3,4
64	H2O2 Sample Return	Outboard	2P33-F013	AO Globe	C	Spring	Air/AC	10	5	Open	Closed	1,2,3,4
66A	Recirc Line A Flow	Inboard	Orifice	-	-	-	-	-	-	-	-	
66A	Recirc Line A Flow	Outboard	2B31-F009A	EFCV	A	Spring	Process	-	-	Open	Open	56
66A	Recirc Line A Flow	Outboard	2B31-F009D	EFCV	A	Spring	Process	-	-	Open	Open	56
66B	Recirc Line A Flow	Inboard	Orifice	-	-	-	-	-	-	-	-	
66B	Recirc Line A Flow	Outboard	2B31-F010A	EFCV	A	Spring	Process	-	-	Open	Open	56
66B	Recirc Line A Flow	Outboard	2B31-F010D	EFCV	A	Spring	Process	-	-	Open	Open	56
67	Drywell Post Accident Vent	Inboard	2T48-F335B	AO Globe	C	Air/AC	Spring	11	4	Closed	Closed	1,2,3,4,12,17

TABLE T7.0-1 (Sheet 14 of 31)
PRIMARY CONTAINMENT ISOLATION DEVICES - TESTABLE PENETRATIONS

PEN. NO.	DESCRIPTION	BARRIER ORIENT.	BARRIER TYPE (MPL)	VALVE TYPE (A)	TEST TYPE (B)	POWER TO OPEN (A), (C)	POWER TO CLOSE (A), (C)	ISOL. GP. (D)	MAX. OPER. TIME (sec)	NORM. POS.	POS. ON ISOL. (E)	NOTES (F)
67	Drywell Post Accident Vent	Outboard	2T48-F334B	AO Globe	C	Air/AC	Spring	11	4	Closed	Closed	1,2,3,4,12,17
69	Drywell/Torus Diff. Press Return	Inboard	2T48-F209	AO Gate	C	Air/AC	Spring	12,k	5	Closed	Closed	1,2,3,4
69	Drywell/Torus Diff. Press Return	Outboard	2T48-F210	AO Gate	C	Air/AC	Spring	12,k	5	Closed	Closed	1,2,3,4
79A	Core Spray DP Inst	Inboard	Orifice	-	-	-	-	-	-	-	-	
79A	Core Spray DP Inst	Outboard	2E21-F018A	EFCV	A	Spring	Process	-	-	Open	Open	56
79E	HPCI Steam Line DP Inst	Inboard	Orifice	-	A	-	-	-	-	-	-	
79E	HPCI Steam Line DP Inst	Outboard	2E41-F024C	EFCV	A	Spring	Process	-	-	Open	Open	56
79F	HPCI Steam Line DP Inst	Inboard	Orifice	-	A	-	-	-	-	-	-	
79F	HPCI Steam Line DP Inst	Outboard	2E41-F024A	EFCV	A	Spring	Process	-	-	Open	Open	56
80	Drywell Post Accident Vent	Inboard	2T48-F335A	AO Globe	C	Air/AC	Spring	11	4	Closed	Closed	1,2,3,4,12,17
80	Drywell Post Accident Vent	Outboard	2T48-F334A	AO Globe	C	Air/AC	Spring	11	4	Closed	Closed	1,2,3,4,12,17
81	Drywell N2 Makeup Inlet	Inboard	2T48-F114	AO Globe	C	N2/AC	Spring	-		Closed	Closed	1,2,3,4,17,20
81	Drywell N2 Makeup Inlet	Outboard	2T48-F113	AO Globe	C	N2/AC	Spring	-		Closed	Closed	1,2,3,4,17,20
82	Drywell Head Flange	Inboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5
82	Drywell Head Flange	Outboard	-	-	-	-	-	-	-	-	-	
83A	RPV Stabilizer Access Hatch	Inboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5,54
83A	RPV Stabilizer Access Hatch	Outboard	-	-	-	-	-	-	-	-	-	
83B	RPV Stabilizer Access Hatch	Inboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5,54
83B	RPV Stabilizer Access Hatch	Outboard	-	-	-	-	-	-	-	-	-	
83C	RPV Stabilizer Access Hatch	Inboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5,54
83C	RPV Stabilizer Access Hatch	Outboard	-	-	-	-	-	-	-	-	-	
83D	RPV Stabilizer Access Hatch	Inboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5,54
83D	RPV Stabilizer Access Hatch	Outboard	-	-	-	-	-	-	-	-	-	
83E	RPV Stabilizer Access Hatch	Inboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5,54
83E	RPV Stabilizer Access Hatch	Outboard	-	-	-	-	-	-	-	-	-	
83F	RPV Stabilizer Access Hatch	Inboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5,54
83F	RPV Stabilizer Access Hatch	Outboard	-	-	-	-	-	-	-	-	-	
83G	RPV Stabilizer Access Hatch	Inboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5,54
83G	RPV Stabilizer Access Hatch	Outboard	-	-	-	-	-	-	-	-	-	
83H	RPV Stabilizer Access Hatch	Inboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5,54
83H	RPV Stabilizer Access Hatch	Outboard	-	-	-	-	-	-	-	-	-	
100A	Neutron Monitoring (Elec. Pen.)	Inboard	Canister	-	B	-	-	-	-	-	-	1,2,3,5,57

TABLE T7.0-1 (Sheet 15 of 31)
PRIMARY CONTAINMENT ISOLATION DEVICES - TESTABLE PENETRATIONS

PEN. NO.	DESCRIPTION	BARRIER ORIENT.	BARRIER TYPE (MPL)	VALVE TYPE (A)	TEST TYPE (B)	POWER TO OPEN (A), (C)	POWER TO CLOSE (A), (C)	ISOL. GP. (D)	MAX. OPER. TIME (sec)	NORM. POS.	POS. ON ISOL. (E)	NOTES (F)
100A	Neutron Monitoring (Elec Pen)	Outboard	-	-	-	-	-	-	-	-	-	
100B	Neutron Monitoring (Elec. Pen)	Inboard	Canister	-	B	-	-	-	-	-	-	1,2,3,5,57
100B	Neutron Monitoring (Elec Pen)	Outboard	-	-	-	-	-	-	-	-	-	
100D	Neutron Monitoring (Elec Pen)	Inboard	Canister	-	B	-	-	-	-	-	-	1,2,3,5,57
100D	Neutron Monitoring (Elec Pen)	Outboard	-	-	-	-	-	-	-	-	-	
100E	Neutron Monitoring (Elec. Pen)	Inboard	Canister	-	B	-	-	-	-	-	-	1,2,3,5,57
100E	Neutron Monitoring (Elec. Pen)	Outboard	-	-	-	-	-	-	-	-	-	
100G/H	Neutron Monitoring (Elec Pen)	Inboard	Canister	-	B	-	-	-	-	-	-	1,2,3,5,57
100G/H	Neutron Monitoring (Elec Pen)	Outboard	-	-	-	-	-	-	-	-	-	
100I/J	Neutron Monitoring (Elec Pen)	Inboard	Canister	-	B	-	-	-	-	-	-	1,2,3,5,57
100I/J	Neutron Monitoring (Elec Pen)	Outboard	-	-	-	-	-	-	-	-	-	
101A	Recirc Pump Power (Elec Pen)	Inboard	Canister	-	B	-	-	-	-	-	-	1,2,3,5,57
101A	Recirc Pump Power (Elec. Pen)	Outboard	-	-	-	-	-	-	-	-	-	
101B	Recirc Pump Power (Elec Pen)	Inboard	Canister	-	B	-	-	-	-	-	-	1,2,3,5,57
101B	Recirc Pump Power (Elec Pen)	Outboard	-	-	-	-	-	-	-	-	-	
101C	Recirc Pump Power (Elec Pen)	Inboard	Canister	-	B	-	-	-	-	-	-	1,2,3,5,57
101C	Recirc Pump Power (Elec Pen)	Outboard	-	-	-	-	-	-	-	-	-	
101D	Recirc Pump Power (Elec. Pen)	Inboard	Canister	-	B	-	-	-	-	-	-	1,2,3,5,57
101D	Recirc Pump Power (Elec. Pen)	Outboard	-	-	-	-	-	-	-	-	-	
101E	Recirc Pump Power (Elec Pen)	Inboard	Canister	-	B	-	-	-	-	-	-	1,2,3,5,57
101E	Recirc Pump Power (Elec. Pen)	Outboard	-	-	-	-	-	-	-	-	-	
101F	Recirc Pump Power (Elec. Pen)	Inboard	Canister	-	B	-	-	-	-	-	-	1,2,3,5,57
101F	Recirc Pump Power (Elec. Pen)	Outboard	-	-	-	-	-	-	-	-	-	
102A	Indication and Control (Elec. Pen)	Inboard	Canister	-	B	-	-	-	-	-	-	1,2,3,5,57
102A	Indication and Control (Elec Pen)	Outboard	-	-	-	-	-	-	-	-	-	
103A	Indication and Control (Elec. Pen)	Inboard	Canister	-	B	-	-	-	-	-	-	1,2,3,5,57
103A	Indication and Control (Elec. Pen)	Outboard	-	-	-	-	-	-	-	-	-	
104A	CRD Rod Pos Indic (Elec Pen)	Inboard	Canister	-	B	-	-	-	-	-	-	1,2,3,5,57
104A	CRD Rod Pos Indic (Elec Pen)	Outboard	-	-	-	-	-	-	-	-	-	
104B	CRD Rod Pos. Indic (Elec. Pen)	Inboard	Canister	-	B	-	-	-	-	-	-	1,2,3,5,57
104B	CRD Rod Pos Indic (Elec. Pen)	Outboard	-	-	-	-	-	-	-	-	-	
104C	CRD Rod Pos Indic (Elec. Pen.)	Inboard	Canister	-	B	-	-	-	-	-	-	1,2,3,5,57

TABLE T7.0-1 (Sheet 16 of 31)
PRIMARY CONTAINMENT ISOLATION DEVICES - TESTABLE PENETRATIONS

PEN. NO.	DESCRIPTION	BARRIER ORIENT.	BARRIER TYPE (MPL)	VALVE TYPE (A)	TEST TYPE (B)	POWER TO OPEN (A), (C)	POWER TO CLOSE (A), (C)	ISOL. GP. (D)	MAX. OPER. TIME (sec)	NORM. POS.	POS. ON ISOL. (E)	NOTES (F)
104C	CRD Rod Pos Indic (Elec. Pen)	Outboard	-	-	-	-	-	-	-	-	-	
104F	CRD Rod Pos. Indic (Elec. Pen)	Inboard	Canister	-	B	-	-	-	-	-	-	1,2,3,5,57
104F	CRD Rod Pos Indic (Elec. Pen.)	Outboard	-	-	-	-	-	-	-	-	-	
104G	CRD Rod Pos Indic (Elec Pen)	Inboard	Canister	-	B	-	-	-	-	-	-	1,2,3,5,57
104G	CRD Rod Pos Indic (Elec Pen)	Outboard	-	-	-	-	-	-	-	-	-	
104H	CRD Rod Pos Indic (Elec Pen)	Inboard	Canister	-	B	-	-	-	-	-	-	1,2,3,5,57
104H	CRD Rod Pos Indic (Elec. Pen)	Outboard	-	-	-	-	-	-	-	-	-	
105A	600V Power (Elec. Pen)	Inboard	Canister	-	B	-	-	-	-	-	-	1,2,3,5,57
105A	600V Power (Elec Pen)	Outboard	-	-	-	-	-	-	-	-	-	
105C	600V Power (Elec Pen)	Inboard	Canister	-	B	-	-	-	-	-	-	1,2,3,5,57
105C	600V Power (Elec Pen)	Outboard	-	-	-	-	-	-	-	-	-	
106A	Thermocouples (Elec. Pen)	Inboard	Canister	-	B	-	-	-	-	-	-	1,2,3,5,57
106A	Thermocouples (Elec Pen)	Outboard	-	-	-	-	-	-	-	-	-	
200A	Torus Access Hatch	Inboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5,54
200A	Torus Access Hatch	Outboard	-	-	-	-	-	-	-	-	-	
200B	Torus Access Hatch	Inboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5,54
200B	Torus Access Hatch	Outboard	-	-	-	-	-	-	-	-	-	
201A	Drywell to Torus Vent Line	Inboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5,14
201A	Drywell to Torus Vent Line	Outboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5,14
201B	Drywell to Torus Vent Line	Inboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5,14
201B	Drywell to Torus Vent Line	Outboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5,14
201C	Drywell to Torus Vent Line	Inboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5,14
201C	Drywell to Torus Vent Line	Outboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5,14
201D	Drywell to Torus Vent Line	Inboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5,14
201D	Drywell to Torus Vent Line	Outboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5,14
201E	Drywell to Torus Vent Line	Inboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5,14
201E	Drywell to Torus Vent Line	Outboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5,14
201F	Drywell to Torus Vent Line	Inboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5,14
201F	Drywell to Torus Vent Line	Outboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5,14
201G	Drywell to Torus Vent Line	Inboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5,14
201G	Drywell to Torus Vent Line	Outboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5,14
201H	Drywell to Torus Vent Line	Inboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5,14

TABLE T7.0-1 (Sheet 17 of 31)
PRIMARY CONTAINMENT ISOLATION DEVICES - TESTABLE PENETRATIONS

PEN. NO.	DESCRIPTION	BARRIER ORIENT.	BARRIER TYPE (MPL)	VALVE TYPE (A)	TEST TYPE (B)	POWER TO OPEN (A), (C)	POWER TO CLOSE (A), (C)	ISOL. GP. (D)	MAX. OPER. TIME (sec)	NORM. POS.	POS. ON ISOL. (E)	NOTES (F)
201H	Drywell to Torus Vent Line	Outboard	Expansion Bellows	-	B	-	-	-	-	-	-	1,2,3,5,14
202	Control and Indic (Elec. Pen.)	Inboard	Cannister	-	B	-	-	-	-	-	-	1,2,3,5,57
202	Control and Indic (Elec Pen)	Outboard	-	-	-	-	-	-	-	-	-	
203	RCIC Pump Suction	Inboard	2E51-F003	AO Btfly	A	Spring	Air/AC	-	-	Open	Open	37
203	RCIC Pump Suction	Outboard	2E51-F031	MO Gate	A	DC	DC	-	-	Closed	Closed	37
204A	RHR Pump Suction	Inboard	2E11-F004A	MO Gate	A	AC	AC	-	-	Open	Open	37
204A	RHR Pump Suction	Inboard	2E11-F030A	Relief	A	-	-	-	-	-	-	16,37
204A	RHR Pump Suction	Outboard	Closed System	-	-	-	-	-	-	-	-	24
204B	RHR Pump Suction	Inboard	2E11-F004B	MO Gate	A	AC	AC	-	-	Open	Open	37
204B	RHR Pump Suction	Inboard	2E11-F030B	Relief	A	-	-	-	-	-	-	16,37
204B	RHR Pump Suction	Outboard	Closed System	-	-	-	-	-	-	-	-	24
204C	RHR Pump Suction	Inboard	2E11-F004C	MO Gate	A	AC	AC	-	-	Open	Open	37
204C	RHR Pump Suction	Inboard	2E11-F030C	Relief	A	-	-	-	-	-	-	16,37
204C	RHR Pump Suction	Outboard	Closed System	-	-	-	-	-	-	-	-	24
204D	RHR Pump Suction	Inboard	2E11-F004D	MO Gate	A	AC	AC	-	-	Open	Open	37
204D	RHR Pump Suction	Inboard	2E11-F030D	Relief	A	-	-	-	-	-	-	16,37
204D	RHR Pump Suction	Outboard	Closed System	-	-	-	-	-	-	-	-	24
205	Vacuum Relief	Inboard	2T48-F310	AO Btfly	C	Spring	Air/AC	-	-	Closed	Closed	1,2,3,4,20
205	Vacuum Relief	Inboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5,46,47
205	Vacuum Relief	Inboard	2T48-F311	AO Btfly	C	Spring	Air/AC	-	-	Closed	Closed	1,2,3,4,20
205	Vacuum Relief	Inboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5,46,47
205	Torus Purge Supply	Inboard	2T48-F309	AO Btfly	C	Air/AC	Spring	2	5	Closed	Closed	1,2,3,4
205	Torus Purge Supply	Inboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5,46,47
205	Torus N2 Makeup	Inboard	2T48-F118B	AO Globe	C	Air/AC	Spring	11	5	Open	Open	1,2,3,4
205	Torus Pressure	Inboard	Orifice	-	-	-	-	-	-	-	-	
205	Vacuum Relief	Outboard	2T48-F328A	AO Check	C	VAC,Air/AC	Reverse Flow	-	-	Closed	Closed	1,2,3,4
205	Vacuum Relief	Outboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5
205	Vacuum Relief	Outboard	2T48-F328B	AO Check	C	VAC,Air/AC	Reverse Flow	-	-	Closed	Closed	1,2,3,4
205	Vacuum Relief	Outboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5
205	Torus Purge Supply	Outboard	2T48-F324	AO Btfly	C	Air/AC	Spring	2	5	Closed	Closed	1,2,3,4
205	Torus Purge Supply	Outboard	2T48-D006	Blind Flange		-	-	-	-	-	-	
205	Torus Purge Supply	Outboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5,47

TABLE T7.0-1 (Sheet 18 of 31)
PRIMARY CONTAINMENT ISOLATION DEVICES - TESTABLE PENETRATIONS

PEN. NO.	DESCRIPTION	BARRIER ORIENT.	BARRIER TYPE (MPL)	VALVE TYPE (A)	TEST TYPE (B)	POWER TO OPEN (A), (C)	POWER TO CLOSE (A), (C)	ISOL. GP. (D)	MAX. OPER. TIME (sec)	NORM. POS.	POS. ON ISOL. (E)	NOTES (F)
205	Torus N2 Makeup	Outboard	2T48-F104	AO Globe	C	Air/AC	Spring	11	5	Closed	Closed	1,2,3,4
205	Torus Pressure	Outboard	2T48-F364B	AO Globe	C	Spring	Air/AC	-	-	Open	Open	1,2,3,4,33
206A	Torus Water Level	Inboard	Orifice	-	-	-	-	-	-	-	-	
206A	Torus Water Level	Outboard	2T48-F361B	AO Globe	C	Spring	Air/AC	-	-	Open	Open	1,2,3,4,33
206C	Torus Water Level	Inboard	Orifice	-	-	-	-	-	-	-	-	
206C	PASS	Inboard	2E41-F122	SO Globe	C	AC	Spring	-	-	Closed	Closed	1,2,3,4,58
206C	Torus Water Level	Outboard	2T48-F361A	AO Globe	C	Spring	Air/AC	-	-	Open	Open	1,2,3,4,33
206C	PASS	Outboard	2E41-F121	SO Globe	C	AC	Spring	-	-	Closed	Closed	1,2,3,4,58
206F	Torus Water Level	Inboard	Orifice	-	-	-	-	-	-	-	-	
206F	Torus Water Level	Outboard	2T48-F362A	AO Globe	A	Spring	Air/AC	-	-	Open	Open	37
206H	Torus Water Level	Inboard	Orifice	-	-	-	-	-	-	-	-	
206H	Torus Water Level	Outboard	2T48-F362B	AO Globe	A	Spring	Air/AC	-	-	Open	Open	37
207	HPCI Pump Suction	Inboard	2E41-F051	AO Bifly	A	Spring	Air/AC		-	Open	Open	37
207	HPCI Pump Suction	Outboard	2E41-F042	MO Gate	A	DC	DC	3	84	Closed	Closed	37
208A	Core Spray Pump Suction	Inboard	2E21-F001A	MO Gate	A	AC	AC	-	-	Open	Open	37
208A	Core Spray Pump Suction	Outboard	Closed System	-	-	-	-	-	-	-	-	24
208B	Core Spray Pump Suction	Inboard	2E21-F001B	MO Gate	A	AC	AC	-	-	Open	Open	37
208B	Core Spray Pump Suction	Outboard	Closed System	-	-	-	-	-	-	-	-	24
210A	RHR Test Line	Inboard	2E11-F025A	Relief	A	-	-	-	-	Closed	Closed	16,37
210A	RHR Test Line	Inboard	2E11-F029	Relief	A	-	-	-	-	Closed	Closed	16,37
210A	RHR Test Line	Inboard	2E11-F097	Relief	A	-	-	-	-	Closed	Closed	16,37
210A	RHR Test Line	Inboard	2E51-F019	MO Globe	A	DC	DC	l	5	Closed	Closed	37
210A	RHR Test Line	Inboard	2E11-F028A	MO Gate	C	AC	AC	g	50	Closed/KL	Closed	1,2,3,4,8,48
210A	RHR Test Line	Outboard	Closed System	-		-	-	-	-	-	-	24
210A	RHR Test Line	Outboard	2E51-F021	Check	A	Process	Reverse Flow	-	-	Closed	Closed	
210B	RHR Test Line	Inboard	2E11-F025B	Relief	A	-	-	-	-	Closed	Closed	16,37
210B	RHR Test Line	Inboard	2E41-F012	MO Globe	A	DC	DC	h	10	Closed	Closed	37
210B	RHR Test Line	Inboard	2E11-F028B	MO Gate	C	AC	AC	g	50	Closed/KL	Closed	1,2,3,4,8,48
210B	RHR Test Line	Outboard	Closed System	-		-	-	-	-	-	-	24
210B	RHR Test Line	Outboard	2E41-F046	Check	A	Process	Reverse Flow	-	-	Closed	Closed	37
211A	Torus Spray	Inboard	2E11-F028A	MO Gate	C	AC	AC	g	50	Closed/KL	Closed	1,2,3,4,8,48
211A	Torus Spray	Outboard	Closed System	-	-	-	-	-	-	-	-	24

TABLE T7.0-1 (Sheet 19 of 31)
PRIMARY CONTAINMENT ISOLATION DEVICES - TESTABLE PENETRATIONS

PEN. NO.	DESCRIPTION	BARRIER ORIENT.	BARRIER TYPE (MPL)	VALVE TYPE (A)	TEST TYPE (B)	POWER TO OPEN (A), (C)	POWER TO CLOSE (A), (C)	ISOL. GP. (D)	MAX. OPER. TIME (sec)	NORM. POS.	POS. ON ISOL. (E)	NOTES (F)
211B	Torus Spray	Inboard	2E11-F028B	MO Gate	C	AC	AC	9	50	Closed/KL	Closed	1,2,3,4,8,48
211B	Torus Spray	Outboard	Closed System	-	-	-	-	-	-	-	-	24
212	RCIC Turbine Exhaust	Inboard	2E51-F001	Stop Check/LO	A	Process	Reverse Flow	-	-	Closed	Closed	37
212	RCIC Turbine Exhaust Vac Brkr	Inboard	2E51-F104	MO Gate	C	AC	AC	9	20	Open	Closed	1,2,3,4,50
212	RCIC Turbine Exhaust	Outboard	2E51-F040	Check	A	Process	Reverse Flow	-	-	Closed	Closed	37
212	RCIC Turbine Exhaust Vac Brkr	Outboard	2E51-F105	MO Gate	C	AC	AC	9	20	Open	Closed	1,2,3,4,50
213	RCIC Turbine Vacuum Pump Disc	Inboard	2E51-F002	Stop Check/LO	A	Process	Reverse Flow	-	-	Closed	Closed	37
213	RCIC Turbine Vacuum Pump Disc	Outboard	2E51-F028	Check	A	Process	Reverse Flow	-	-	Closed	Closed	37
214	HPCI Turbine Exhaust	Inboard	2E41-F021	Stop Check/LO	A	Process	Reverse Flow	-	-	Closed	Closed	37
214	HPCI Turbine Exhaust Vac Brkr	Inboard	2E41-F104	MO Gate	C	AC	AC	8	20	Open	Closed	1,2,3,4,51
214	HPCI Turbine Exhaust	Outboard	2E41-F049	Check	A	Process	Reverse Flow	-	-	Closed	Closed	37
214	HPCI Turbine Exhaust Vac Brkr	Outboard	2E41-F111	MO Gate	C	AC	AC	8	20	Open	Closed	1,2,3,4,51
215	HPCI Exhaust Drain	Inboard	2E41-F022	Stop Check/LO	A	Process	Reverse Flow	-		Closed	Closed	37
215	HPCI Exhaust Drain	Outboard	2E41-F040	Check	A	Process	Reverse Flow	-		Closed	Closed	37
217A	H2O2 Sample Supply	Inboard	2P33-F006	AO Globe	C	Spring	Air/DC	10	5	Open	Closed	1,2,3,4
217A	H2O2 Sample Supply	Outboard	2P33-F014	AO Globe	C	Spring	Air/AC	10	5	Open	Closed	1,2,3,4
217B	H2O2 Sample Supply	Inboard	2P33-F007	AO Globe	C	Spring	Air/DC	10	5	Open	Closed	1,2,3,4
217B	H2O2 Sample Supply	Outboard	2P33-F015	AO Globe	C	Spring	Air/AC	10	5	Open	Closed	1,2,3,4
217C	FPM Sample Supply	Inboard	2D11-F065	Globe	C	Hand	Hand	-	-	Closed	Closed	1,2,3,4,18
217C	FPM Sample Supply	Outboard	2D11-F058	Globe	C	Hand	Hand		-	Closed	Closed	1,2,3,4,18
218A	Torus Purification Suction	Inboard	2G51-F002	Gate	A	Hand	Hand	-	-	Closed/LC	Closed	18,37
218A	Torus Purification Suction	Inboard	Flange Gasket	-	A	-	-	-	-	-	-	37
218A	Torus Purification Suction	Outboard	2G51-D001	Blind Flange	A	-	-	-	-	-	-	37
218B	Construction Drain	Inboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5,37,54
218B	Construction Drain	Outboard	-	-	-	-	-	-	-	-	-	
220	Torus Exhaust Bypass	Inboard	2T48-F339	AO Globe	C	Air/AC	Spring	10	5	Closed	Closed	1,2,3,4
220	Torus Main Exhaust	Inboard	2T48-F318	AO Bfly	C	Air/AC	Spring	2	5	Closed	Closed	1,2,3,4
220	Torus Main Exhaust	Inboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5,52
220	Torus Pressure	Inboard	Orifice	-	-	-	-	-	-	-	-	
220	Torus Exhaust Bypass	Outboard	2T48-F338	AO Globe	C	Air/AC	Spring	10	5	Closed	Closed	1,2,3,4
220	Torus Main Exhaust	Outboard	2T48-F326	AO Bfly	C	Air/AC	Spring	2	5	Closed	Closed	1,2,3,4
220	Torus Main Exhaust	Outboard	Double O-Ring	-	C	-	-	-	-	-	-	1,2,3,5,53

TABLE T7.0-1 (Sheet 20 of 31)
PRIMARY CONTAINMENT ISOLATION DEVICES - TESTABLE PENETRATIONS

PEN. NO.	DESCRIPTION	BARRIER ORIENT.	BARRIER TYPE (MPL)	VALVE TYPE (A)	TEST TYPE (B)	POWER TO OPEN (A), (C)	POWER TO CLOSE (A), (C)	ISOL. GP. (D)	MAX. OPER. TIME (sec)	NORM. POS.	POS. ON ISOL. (E)	NOTES (F)
220	Torus Pressure	Outboard	2T48-F364A	AO Globe	C	Spring	Air/AC	-	-	Open	Open	1,2,3,4,33
221A	Post LOCA H2 Recomb Return A	Inboard	2T49-F004A	MO Gate	C	AC	AC	-	-	Closed/KL	Closed	1,2,3,4,17
221A	Post LOCA H2 Recomb Return A	Outboard	Closed System	-	C	-	-	-	-	-	-	31
221B	HPCI Turbine Exhaust Vac Brkr	Inboard	2E41-F111	MO Gate	C	AC	AC	8	20	Open	Closed	1,2,3,4
221B	HPCI Turbine Exhaust Vac Brkr	Outboard	2E41-F104	MO Gate	C	AC	AC	8	20	Open	Closed	1,2,3,4
221C	RCIC Turbine Exhaust Vac Brkr	Inboard	2E51-F105	MO Gate	C	AC	AC	9	20	Open	Closed	1,2,3,4
221C	RCIC Turbine Exhaust Vac Brkr	Outboard	2E51-F104	MO Gate	C	AC	AC	9	20	Open	Closed	1,2,3,4
222B	Post LOCA H2 Recomb Return B	Inboard	2T49-F004B	MO Gate	C	AC	AC	-	-	Closed/KL	Closed	1,2,3,4
222B	Post LOCA H2 Recomb Return B	Outboard	Closed System	-	C	-	-	-	-	-	-	31
223A	Spare	Inboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5
223A	Spare	Outboard	-	-	-	-	-	-	-	-	-	
223B	Spare	Inboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5
223B	Spare	Outboard	-	-	-	-	-	-	-	-	-	
224A	RHR Relief Valve Disch (RHR)	Inboard	2E11-F055A	Relief	A	-	-	-	-	Closed	-	16,37
224A	RHR Relief Valve Disch (RHR)	Inboard	2E11-F103A	MO Globe	A	AC	AC	-	-	Closed	Closed	17,37
224A	RHR Relief Valve Disch (RHR)	Inboard	2E11-F3078A	Relief	A	-	-	-	-	Closed	-	16,37
224A	Hydrogen Recombiner	Inboard	2T49-F009A	Relief	A	-	-	-	-	Closed	-	16,37
224A	RHR Relief Valve Disch (RHR)	Outboard	Closed System	-	-	-	-	-	-	-	-	24
224A	Hydrogen Recombiner	Outboard	Closed System	-	-	-	-	-	-	-	-	24,31
224B	RHR Relief Valve Disch (RHR)	Inboard	2E11-F055B	Relief	A	-	-	-	-	Closed	-	16,37
224B	RHR Relief Valve Disch (RHR)	Inboard	2E11-F103B	MO Globe	A	AC	AC	-	-	Closed	Closed	17,37
224B	RHR Relief Valve Disch (RHR)	Inboard	2E11-F3078B	Relief	A	-	-	-	-	Closed	-	16,37
224B	Hydrogen Recombiner	Inboard	2T49-F009B	Relief	A	-	-	-	-	Closed	-	16,37
224B	RHR Relief Valve Disch (RHR)	Outboard	Closed System	-	-	-	-	-	-	-	-	24
224B	Hydrogen Recombiner	Outboard	Closed System	-	-	-	-	-	-	-	-	24,31
225A	Vac Breaker Air Supply	Inboard	2T48-F323G Air Cyl	-	C	-	-	-	-	-	-	38
225A	Vac Breaker Air Supply	Outboard	2T48-F342G	SO Globe	C	AC	Spring	-	-	Closed	Closed	1,2,3,4
225B	Vac Breaker Air Supply	Inboard	2T48-F323H Air Cyl	-	C	-	-	-	-	-	-	38
225B	Vac Breaker Air Supply	Outboard	2T48-F342H	SO Globe	C	AC	Spring	-	-	Closed	Closed	1,2,3,4
225C	Vac Breaker Air Supply	Inboard	2T48-F323I Air Cyl	-	C	-	-	-	-	-	-	38
225C	Vac Breaker Air Supply	Outboard	2T48-F342I	SO Globe	C	AC	Spring	-	-	Closed	Closed	1,2,3,4
225D	Vac Breaker Air Supply	Inboard	2T48-F323J Air Cyl.	-	C	-	-	-	-	-	-	38

TABLE T7.0-1 (Sheet 21 of 31)
PRIMARY CONTAINMENT ISOLATION DEVICES - TESTABLE PENETRATIONS

PEN. NO.	DESCRIPTION	BARRIER ORIENT.	BARRIER TYPE (MPL)	VALVE TYPE (A)	TEST TYPE (B)	POWER TO OPEN (A), (C)	POWER TO CLOSE (A), (C)	ISOL. GP. (D)	MAX. OPER. TIME (sec)	NORM. POS.	POS. ON ISOL. (E)	NOTES (F)
225D	Vac Breaker Air Supply	Outboard	2T48-F342J	SO Globe	C	AC	Spring	-	-	Closed	Closed	1,2,3,4
225E	Vac Breaker Air Supply	Inboard	2T48-F323K Air Cyl	-	C	-	-	-	-	-	-	38
225E	Vac Breaker Air Supply	Outboard	2T48-F342K	SO Globe	C	AC	Spring	-	-	Closed	Closed	1,2,3,4
225F	Vac Breaker Air Supply	Inboard	2T48-F323L Air Cyl	-	C	-	-	-	-	-	-	38
225F	Vac Breaker Air Supply	Outboard	2T48-F342L	SO Globe	C	AC	Spring	-	-	Closed	Closed	1,2,3,4
225G	Vac Breaker Air Supply	Inboard	2T48-F323A Air Cyl	-	C	-	-	-	-	-	-	38
225G	Vac Breaker Air Supply	Outboard	2T48-F342A	SO Globe	C	AC	Spring	-	-	Closed	Closed	1,2,3,4
225H	Vac Breaker Air Supply	Inboard	2T48-F323B Air Cyl	-	C	-	-	-	-	-	-	38
225H	Vac Breaker Air Supply	Outboard	2T48-F342B	SO Globe	C	AC	Spring	-	-	Closed	Closed	1,2,3,4
225J	Vac Breaker Air Supply	Inboard	2T48-F323CAir Cyl	-	C	-	-	-	-	-	-	38
225J	Vac Breaker Air Supply	Outboard	2T48-F342C	SO Globe	C	AC	Spring	-	-	Closed	Closed	1,2,3,4
225K	Vac Breaker Air Supply	Inboard	2T48-F323D Air Cyl	-	C	-	-	-	-	-	-	38
225K	Vac Breaker Air Supply	Outboard	2T48-F342D	SO Globe	C	AC	Spring	-	-	Closed	Closed	1,2,3,4
225L	Vac Breaker Air Supply	Inboard	2T48-F323E Air Cyl	-	C	-	-	-	-	-	-	38
225L	Vac Breaker Air Supply	Outboard	2T48-F342E	SO Globe	C	AC	Spring	-	-	Closed	Closed	1,2,3,4
225M	Vac Breaker Air Supply	Inboard	2T48-F323F Air Cyl	-	C	-	-	-	-	-	-	38
225M	Vac Breaker Air Supply	Outboard	2T48-F342F	SO Globe	C	AC	Spring	-	-	Closed	Closed	1,2,3,4
226A	Core Spray Test Line	Inboard	2E21-F015A	MO Globe	A	AC	AC	f	57	Closed	Closed	37
226A	Core Spray Test Line	Inboard	2E21-F036A	Check	A	Process	Reverse Flow	-	-	Closed	Closed	37
226A	Core Spray Test Line	Inboard	2E21-F044A	Stop Check	A	Process	Reverse Flow	-	-	Open	Closed	37
226A	Core Spray Test Line	Inboard	2E11-F011A	MO Gate	A	AC	AC	g	24	Closed	Closed	37
226A	Core Spray Test Line	Inboard	2E11-F007A	MO Gate	A	AC	AC	-	-	Open	Closed	22,37
226A	Core Spray Test Line	Outboard	2E11-F026A	MO Gate	A	AC	AC	g	24	Closed	Closed	37
226A	Core Spray Test Line	Outboard	Closed System	-	-	-	-	-	-	-	-	24
226B	Core Spray Test Line	Inboard	2E21-F015B	MO Globe	A	AC	AC	f	57	Closed	Closed	37
226B	Core Spray Test Line	Inboard	2E21-F036B	Check	A	Process	Reverse Flow	-	-	Closed	Closed	37
226B	Core Spray Test Line	Inboard	2E21-F044B	Stop Check	A	Process	Reverse Flow	-	-	Open	Closed	37
226B	Core Spray Test Line	Inboard	2E11-F011B	MO Gate	A	AC	AC	g	24	Closed	Closed	37
226B	Core Spray Test Line	Inboard	2E11-F007B	MO Gate	A	AC	AC	-	-	Open	Closed	22,37
226B	Core Spray Test Line	Outboard	Closed System	-	-	-	-	-	-	-	-	24
226B	Core Spray Test Line	Outboard	2E11-F026B	MO Gate	A	AC	AC	g	24	Closed	Closed	37
228A	Low Voltage Power	Inboard	Cannister	-	B	-	-	-	-	-	-	1,2,3,5,57

TABLE T7.0-1 (Sheet 22 of 31)
PRIMARY CONTAINMENT ISOLATION DEVICES - TESTABLE PENETRATIONS

PEN. NO.	DESCRIPTION	BARRIER ORIENT.	BARRIER TYPE (MPL)	VALVE TYPE (A)	TEST TYPE (B)	POWER TO OPEN (A), (C)	POWER TO CLOSE (A), (C)	ISOL. GP. (D)	MAX. OPER. TIME (sec)	NORM. POS.	POS. ON ISOL. (E)	NOTES (F)
228A	Low Voltage Power	Outboard	-	-	-	-	-	-	-	-	-	
228B	Low Voltage Power (spare)	Inboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5
228B	Low Voltage Power (spare)	Outboard	-	-	-	-	-	-	-	-	-	
228C	Low Voltage Power	Inboard	Cannister	-	B	-	-	-	-	-	-	1,2,3,5,57
228C	Low Voltage Power	Outboard	-	-	-	-	-	-	-	-	-	
230	Torus N2 Makeup Inlet	Inboard	2T48-F327	AO Globe	C	N2/AC	Spring	-	-	Closed	Closed	1,2,3,4,17,20
230	Torus N2 Makeup Inlet	Outboard	2T48-F325	AO Globe	C	N2/AC	Spring	-	-	Closed	Closed	1,2,3,4,17,20
231	Torus Post Accident Vent	Inboard	2T48-F333A	AO Globe	C	Air/AC	Spring	10	4	Closed	Closed	1,2,3,4,12,17,20
231	Torus Post Accident Vent	Outboard	2T48-F332A	AO Globe	C	Air/AC	Spring	10	4	Closed	Closed	1,2,3,4,12,17,20
233	Drywell to Torus DP Suction	Inboard	2T48-F211	AO Globe	C	Air/AC	Spring	12,k	5	Closed	Closed	1,2,3,4
233	Drywell to Torus DP Suction	Outboard	2T48-F212	AO Gate	C	Air/AC	Spring	12,k	5	Closed	Closed	1,2,3,4
234A	Cond Pump Suction from Torus	Inboard	2G51-F011	AO Globe	A	Air/AC	Spring	7	15	Open	Closed	37
234A	Cond Pump Suction from Torus	Outboard	2G51-F012	AO Globe	A	Air/AC	Spring	7	15	Open	Closed	37
235A	Torus Post Accident Vent	Inboard	2T48-F333B	AO Globe	C	Air/AC	Spring	10	4	Closed	Closed	1,2,3,4,12,17,20
235A	Torus Post Accident Vent	Outboard	2T48-F332B	AO Globe	C	Air/AC	Spring	10	4	Closed	Closed	1,2,3,4,12,17,20
235B	Torus N2 Makeup Inlet	Inboard	2T48-F116	AO Globe	C	N2/AC	Spring	-	-	Closed	Closed	1,2,3,4,17,20
235B	Torus N2 Makeup Inlet	Outboard	2T48-F115	AO Globe	C	N2/AC	Spring	-	-	Closed	Closed	1,2,3,4,17,20
236	Torus Access Hatch	Inboard	Double O-Ring	-	B	-	-	-	-	-	-	1,2,3,5,54
236	Torus Access Hatch	Outboard	-	-	-	-	-	-	-	-	-	

TABLE T7.0-1 (Sheet 23 of 31)
PRIMARY CONTAINMENT ISOLATION DEVICES - TESTABLE PENETRATIONS

<u>NOTES</u>

A. All motor operated isolation valves remain in the last position upon failure of valve power.

B. Although specific penetrations are listed as receiving type A, B, or C tests, only those penetrations that do not get a type B or C test are listed as getting a Type A test.

C. The AC motor operated valves are powered from the AC standby emergency buses. The DC powered isolation valves are powered from the plant batteries.

D. <u>Isolation groups for automatic PCIVs are defined as follows:</u>

<u>GROUP 1</u>: The valves in Group 1 are actuated by any <u>one</u> of the following conditions:

1. Reactor vessel water level - Low Low Low, Level 1
2. Main steam line flow - High
3. Main steam line tunnel temperature - High
4. Main steam line pressure - Low
5. Condenser vacuum - Low
6. Turbine building area temperature - High

<u>GROUP 2</u>: The valves in Group 2 are actuated by any <u>one</u> of the following conditions:

1. Reactor vessel water level - Low, Level 3
2. Drywell pressure - High
3. Drywell radiation - High*
4. Reactor building exhaust radiation - High*
5. Refueling floor exhaust radiation - High*

* This signal isolates the 18 inch containment purge and vent valves only.

TABLE T7.0-1 (Sheet 24 of 31)
PRIMARY CONTAINMENT ISOLATION DEVICES - TESTABLE PENETRATIONS

D. GROUP 3: The valves in Group 3 are actuated by any one of the following conditions:

1. HPCI steam line flow - High
2. HPCI steam supply pressure - Low
3. HPCI turbine exhaust diaphragm pressure - High
4. Suppression pool area ambient temperature - High**
5. Suppression pool area differential temperature - High**
6. Suppression pool area temperature - Time Delay Relays
7. Emergency area cooler temperature - High
8. HPCI pipe penetration room temperature - High

** This signal must be present for more than 15 minutes before system isolation will take place via the suppression pool area temperature-time delay relays.

GROUP 4: The valves in Group 4 are actuated by any one of the following conditions:

1. RCIC steam line flow - High
2. RCIC steam line pressure - Low
3. RCIC turbine exhaust diaphragm pressure - High
4. RCIC suppression pool area ambient temperature - High***
5. RCIC suppression pool area differential temperature - High***
6. RCIC suppression pool area temperature - Time Delay Relays
7. Emergency area cooler temperature - High

*** This signal must be present for more than 30 minutes before system isolation will take place via the suppression pool area temperature-time delay relays.

GROUP 5: The valves in Group 5 are actuated by any one of the following conditions:

1. Reactor vessel water level - Low Low, Level 2
2. Reactor water cleanup area temperature - High
3. Reactor water cleanup area ventilation differential temperature - High
4. Standby Liquid Control System initiation****

**** Closes 2G31-F004 only

TABLE T7.0-1 (Sheet 25 of 31)
PRIMARY CONTAINMENT ISOLATION DEVICES - TESTABLE PENETRATIONS

D. <u>GROUP 6</u>: The valves in Group 6 are actuated by any <u>one</u> of the following conditions:

1. Reactor vessel water level - Low, Level 3
2. Reactor steam dome pressure - High

<u>GROUP 7</u>: The valves in Group 7 are actuated by any <u>one</u> of the following conditions:

1. Drywell pressure high
2. Reactor vessel water level - Low, Level 3
3. Reactor building exhaust radiation high
4. Refueling floor exhaust radiation high

<u>GROUP 8</u>: The valves in Group 8 are actuated by concurrent receipt of the following signals:

1. HPCI steam supply pressure - Low
2. Drywell pressure - High

<u>GROUP 9</u>: The valves in Group 9 are actuated by concurrent receipt of the following signals:

1. RCIC steam line pressure - Low
2. Drywell pressure - High

<u>GROUP 10</u>: The valves in Group 10 are actuated by any <u>one</u> of the following conditions:

1. Reactor vessel water level - Low, Level 3
2. Drywell pressure - High
3. Reactor building exhaust radiation - High
4. Refueling floor exhaust radiation - High

<u>GROUP 11</u>: The valves in Group 11 are actuated by any <u>one</u> of the following conditions:

1. Reactor vessel water level - Low, Level 3
2. Drywell pressure - High
3. Reactor building exhaust radiation - High
4. Refueling floor exhaust radiation - High

D. <u>GROUP 12</u>: The valves in Group 12 are actuated by any <u>one</u> of the following conditions:

1. Reactor vessel water level - Low, Level 3
2. Drywell pressure - High
3. Reactor building exhaust radiation - High
4. Refueling floor exhaust radiation - High

<u>Other isolation signal designators</u>:

a. 2B31-F019 and 2B31-F020 also isolate on main steam line radiation - high, high.

b. These valves do <u>NOT</u> isolate on reactor building exhaust radiation - high or refueling floor exhaust radiation - high or drywell radiation - high signals.

c. These valves isolate on high flow in drywell pneumatic supply line signal.

d. These valves also isolate on RWCU differential flow - high. 2G31-F004 also isolates on high temperature following the non-regenerative heat exchanger.

e. These valves close upon withdrawal of the TIP. TIP automatic withdrawal is actuated by either reactor vessel water level - low or drywell pressure - high.

f. These valves isolate on Core Spray actuation via a reactor vessel water level - low low low, level 1 or drywell pressure - high signal.

g. These valves isolate on LPCI actuation via a reactor vessel water level - low low low, level 1 or drywell pressure - high signal.

h. This valve closes when the HPCI steam supply valve or the HPCI turbine stop valve is closed or on HPCI pump discharge flow - high.

i. This valve closes when the RCIC steam supply valve or the RCIC turbine stop valve is closed or on RCIC pump discharge flow - high.

j. These valves automatically isolate under the following conditions: 1) 2E11-F008 not closed <u>AND</u> 2) 2E11-F009 not closed <u>and</u> 3) reactor pressure ≤ 145 psig <u>AND</u> 4) high drywell radiation <u>OR</u> Reactor vessel water level - Low, Level 3.

k. These valves also isolate on main steam line radiation - high, high.

PRIMARY CONTAINMENT ISOLATION DEVICES - TESTABLE PENETRATIONS

E. The Position on Isolation results from the listed Normal Position receiving an isolation signal.

F. NOTES:

1. All type C test durations will generally exceed 1 hour.

2. Test pressures are at least 48.7 psig for all valves and penetrations except MSIVs which are tested at 28.8 psig.

3. The total acceptable leakage for all valves and penetrations other than the MSIVs is 0.6 L_a.

4. Local leak tests on all testable isolation valves shall be performed in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.

5. Local leak tests on all testable penetrations shall be performed in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.

6. The personnel airlock shall be tested in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.

7. MSIVs require that both solenoid pilots be de-energized to close valves. The accumulator air pressure, plus spring, act together to close valves when both pilots are de energized. Voltage failure at only one pilot does not cause valve closure. The valves are designed to close fully in less than 5 seconds, but in no case less than 3 seconds.

8. Containment spray and suppression cooling valves have interlocks that allow them to be reopened manually after automatic closure. This setup permits containment spray for high drywell pressure conditions and/or suppression water cooling. When automatic signals are not present, these valves may be opened for testing or operating convenience.

9. These valves undergo a water test. Leakage rate must not exceed a value that would deplete the water inventory covering the valve seating surface in a 30-day period. Measured local leakage is not added to the types B and C local leakage totals for comparison with the 60-percent L_a acceptance criteria for local leakage rate testing.

10. Control rod hydraulic lines can be isolated by the solenoid valves outside the primary containment. Lines that extend outside the primary containment are small and terminate in a system designed to prevent out leakage. Solenoid valves normally are closed, but they open on rod movement and during a reactor scram.

PRIMARY CONTAINMENT ISOLATION DEVICES - TESTABLE PENETRATIONS

F.

11. Coincident signals (RPV Water Level Low, Low, Low - Level 1 + Drywell Pressure High) and a low reactor pressure permissive open valves. Special interlocks permit testing these valves by a manual switch except when automatic isolation signals are present.

12. Manual switches override all automatic signals on the two smaller valves that bypass the suppression chamber and drywell exhaust valve.

13. Signals RPV Water Level Low - Level 3 or High Drywell Pressure cause automatic withdrawal of the traversing incore probe. When the probe is withdrawn, the valve automatically closes by mechanical action. An explosive shear valve is installed outboard of the ball valve. The shear valve is provided to isolate the line if the probe does not withdraw.

14. There is one bellows assembly on each torus downcomer from the drywell to the torus. The drywell penetrations are X-5A-H and the torus penetrations are X-201A-H. Although the same bellows assemblies are listed under both penetration numbers in these tables for completeness, they are listed only under X-5A-H in the LLRT procedure for simplicity.

15. (DELETED)

16. Relief valve setpoint is greater than 1.5 times the containment design pressure. Relief valve discharge side serves as a boundary.

17. Administratively closed.

18. Locked-closed manual valve.

19. Leakage detection is provided by process instrumentation.

20. Alarms in control room when valve is open.

21. (DELETED)

22. Valve will close after RHR flow is established.

23. (DELETED)

24. The second isolation boundary is provided by a Quality Group B, Seismic Category I, missile protected, closed system. The system is filled with water and operating at a pressure greater than P_a, post-LOCA.

F.

25. Isolation of the drywell pneumatic system is provided by flow instrumentation which will generate a high-flow signal and automatically close the redundant header isolation valves (after a time delay) should the pneumatic header be ruptured in the drywell. In this case, the redundant pneumatic header will provide the double barrier required for containment isolation.

26. Deactivated and locked in the closed position.

27. The leakage rate through each MSIV shall be ≤ 100 scfh with a combined maximum pathway leakage ≤ 250 scfh for all four main steam lines when tested at ≥ 28.8 psig. The leakage rate acceptance criteria for the first test following discovery of leakage through an MSIV not meeting the 100 scfh limit shall be ≤ 11.5 scfh for that MSIV.

28. The combined leakage rate for penetrations 8, 14, 18, 19, and 55 shall not exceed 0.009 L_a.

29. System remains water filled post-LOCA. Isolation valves are tested with water. Leakage is not included in the 0.6 L_a types B and C tests local leakage totals.

30. These valves are required for limiting containment leakage following a design bases accident and are expected to remain covered by water following a design basis LOCA.

31. The outboard isolation barrier is a closed system outside primary containment.

32. Instrument removed as part of ATTS modification.

33. Seismic Category I, Quality Group B instrument line.

34. Since the TIP drive shear valve isolates the TIP tubing by shearing the tube and drive cable and by jamming the sheared ends of the tubing into a Teflon coating on the shear valve disc, the valve cannot be type C tested without destroying the drive tube. Therefore, the TIP shear valves are not type C tested.

35. The design of these lines does not facilitate type C testing as described in 10 CFR 50, Appendix J. However, adequate leakage monitoring of the CRD lines is provided by normal plant operating procedures. Since the insert and withdraw lines are pressurized to at least reactor operating pressure by the cooling water flow during normal plant operation, leakage from these lines would be immediately evident.

The hydraulic control units are installed on El. 130 ft. of the reactor building, a relatively high traffic area. In addition, the Unit 2 daily rounds procedure requires the operator to make a visual inspection for leakage in the hydraulic area of the reactor building at least once per shift and record the inspection.

F.

36. The inboard isolation barrier is a closed system inside primary containment.

37. This penetration is sealed from the primary containment, and not leakage tested, due to its line terminating below the water level of the torus. No leakage testing is necessary because the torus is postulated to always remain filled with water.

38. The inboard isolation barrier is the vacuum breaker exercising cylinder. The barrier is provided by seals on the air operated piston. The exercising cylinder, although not Quality Group B, was specified by the vacuum breaker vendor to be qualified to the postulated post LOCA environment. The cylinder is designed to operate with an air pressure of 95 to 100 psig, which is significantly higher than the post LOCA containment pressure, and is type C leakage rate tested.

39. The first flange double o-rings on 2T48-F307 act as an inboard barrier for penetration X-25.

40. The second flange double o-rings and shaft double o-rings on valve 2T48-F307, in conjunction with the first flange double o-rings on valves 2T48-F308 and F103, are outboard barriers for penetration X-25.

41. The first flange double o-rings on 2T48-F319 act as an inboard barrier for penetration X-26.

42. The second flange double o-rings and shaft double o-rings on valve 2T48-F319, in conjunction with the first flange double o-rings on valve 2T48-F320, are outboard barriers for penetration X-26.

43. The two check valves used as inboard and outboard barriers have been evaluated to provide sufficient isolation capability. The evaluation was done considering the consequences of breaking the line that these valves are a part of. Furthermore, it was concluded that the installation of an automatic power actuated valve outside primary containment could possibly result in a breach of the primary coolant boundary during normal reactor operation.

44. Although a check valve is not normally used as an outboard barrier, this situation has been evaluated with the conclusion that an automatic valve that opens on signal introduces a possible failure mechanism. For this case an explosive valve is used to provide assurance for reliable timely actuation; therefore, the availability of the line is assured.

45. 2B21-F111 and F112 are outside of the containment boundary. They are type C tested since they will be used post-LOCA to obtain samples.

46. The first flange double o-rings on valves 2T48-F310, F311, and F309 act as an inboard barrier for penetration X-205.

47. The second flange double o-rings and shaft double o-rings on valves 2T48-F310, F311, and F309, in conjunction with the first flange double o-rings on valve 2T48-F324, are outboard barriers for penetration X-205.

PRIMARY CONTAINMENT ISOLATION DEVICES - TESTABLE PENETRATIONS

F. 48. 2E11-F028A&B are tested as barriers for penetrations X-211A & B. Since the penetrations X-210 A & B have water seals, these valves are not required to be tested for these penetrations.

49. (DELETED)

50. Valves 2E51-F104 & F105 are type C tested as part of penetration X-221C.

51. Valves 2E41-F104 & F111 are type C tested as part of penetration X-221B.

52. The first flange double o-rings on 2T48-F318 act as an inboard barrier for penetration X-220.

53. The second flange double o-rings and shaft double o-rings on valve 2T48-F318, in conjunction with the first flange double o-rings on valves 2T48-F326, are outboard barriers for penetration X-220.

54. Penetration is sealed by a blind flange or door with double o-ring seals. These seals are leakage rate tested by pressurizing between the o-rings.

55. The personnel airlock door seals are tested at 10 psig; the barrel is tested at P_a. The lock barrel test leakage rate does not exceed 0.05 L_a.

56. Seismic Category I, Quality Group A, instrument line up to and including the excess flow check valve (EFCV). Instrument tubing is certified and Seismic Category I. An orifice is installed in proximity to the process line in accordance with Regulatory Guide 1.11.

57. Electrical penetrations are tested by pressurizing between the seals through a valved test connection.

58. 2E41-F121 and F122 are outside the containment boundary. They are type C tested since they will be used Post LOCA to obtain samples.

TABLE T7.0-2 (Sheet 1 of 4)
PRIMARY CONTAINMENT ISOLATION DEVICES: MPL TO PENETRATION NUMBER CROSS REFERENCE

MPL NUMBER	PEN. NUMBER	MPL NUMBER	PEN. NUMBER	MPL NUMBER	PEN. NUMBER
2B21-F010A	9A	2B21-F059D	49B	2B31-F003A	31D
2B21-F010B	9B	2B21-F059E	40B (D)	2B31-F003B	30D
2B21-F016	8	2B21-F059F	49D	2B31-F004A	31E
2B21-F019	8	2B21-F059G	40B (E)	2B31-F004B	30E
2B21-F022A	7A	2B21-F059H	49E	2B31-F009A	66A
2B21-F022B	7B	2B21-F059L	45B	2B31-F009B	27A
2B21-F022C	7C	2B21-F059M	50B	2B31-F009C	27A
2B21-F022D	7D	2B21-F059N	45F	2B31-F009D	66A
2B21-F028A	7A	2B21-F059P	50F	2B31-F010A	66B
2B21-F028B	7B	2B21-F059R	45C	2B31-F010B	27B
2B21-F028C	7C	2B21-F059S	50C	2B31-F010C	27B
2B21-F028D	7D	2B21-F059T	45E	2B31-F010D	66B
2B21-F041	56F	2B21-F059U	50E	2B31-F011A	29A
2B21-F043A	56E	2B21-F061	40A (D)	2B31-F011B	57A
2B21-F043B	59E	2B21-F070A	52D	2B31-F011C	57A
2B21-F045A	56C	2B21-F070B	52E	2B31-F011D	29A
2B21-F045B	59C	2B21-F070C	52C	2B31-F012A	29B
2B21-F047A	56B	2B21-F070D	51E	2B31-F012B	57B
2B21-F047B	59B	2B21-F071A	33D	2B31-F012C	57B
2B21-F049A	56D	2B21-F071B	33C	2B31-F012D	29B
2B21-F049B	59D	2B21-F071C	33E	2B31-F013A	27C
2B21-F051A	40B (C)	2B21-F071D	34B	2B31-F013B	57C
2B21-F051B	49C	2B21-F072A	33A	2B31-F017A	27C
2B21-F051C	45D	2B21-F072B	33B	2B31-F017B	57C
2B21-F051D	50D	2B21-F072C	33F	2B31-F019	41
2B21-F053A	40B (A)	2B21-F072D	34A	2B31-F020	41
2B21-F053B	49A	2B21-F073A	52A	2B31-F040A	31B
2B21-F053C	45A	2B21-F073B	52F	2B31-F040B	30B
2B21-F053D	50A	2B21-F073C	52B	2B31-F040C	31C
2B21-F055	40C (C)	2B21-F073D	51F	2B31-F040D	30C
2B21-F057	40C (D)	2B21-F077A	9A	2B31-F058A	31A
2B21-F059A	40B (F)	2B21-F077B	9B	2B31-F058B	30A
2B21-F059B	49F	2B21-F111	45D	2C11-D001-120	37A
2B21-F059C	40B (B)	2B21-F112	45D	2C11-D001-120	37B

TABLE T7.0-2 (Sheet 2 of 4)
PRIMARY CONTAINMENT ISOLATION DEVICES: MPL TO PENETRATION NUMBER CROSS REFERENCE

MPL NUMBER	PEN. NUMBER
2C11-D001-120	37C
2C11-D001-120	37D
2C11-D001-121	38A
2C11-D001-121	38B
2C11-D001-121	38C
2C11-D001-121	38D
2C11-D001-122	38A
2C11-D001-122	38B
2C11-D001-122	38C
2C11-D001-122	38D
2C11-D001-123	37A
2C11-D001-123	37B
2C11-D001-123	37C
2C11-D001-123	37D
2C11-D001-126	37A
2C11-D001-126	37B
2C11-D001-126	37C
2C11-D001-126	37D
2C11-D001-127	38A
2C11-D001-127	38B
2C11-D001-127	38C
2C11-D001-127	38D
2C11-D001-138	37A
2C11-D001-138	37B
2C11-D001-138	37C
2C11-D001-138	37D
2C41-F006	42
2C41-F007	42
2C51-F3012	35E
2C51-F3017	35E
2C51-R751	35E
2C51-R752	35E
2D11-F050	60B
2D11-F051	62

MPL NUMBER	PEN. NUMBER
2D11-F052	60B
2D11-F053	62
2D11-F058	217C
2D11-F065	217C
2E11-F004A	204A
2E11-F004B	204B
2E11-F004C	204C
2E11-F004D	204D
2E11-F007A	226A
2E11-F007B	226B
2E11-F008	12
2E11-F011A	226A
2E11-F011B	226B
2E11-F015A	13A
2E11-F015B	13B
2E11-F016A	39A
2E11-F016B	39B
2E11-F023	17
2E11-F025A	210A
2E11-F025B	210B
2E11-F026A	226A
2E11-F026B	226B
2E11-F028A	210A
2E11-F028A	211A
2E11-F028B	210B
2E11-F028B	211B
2E11-F029	210A
2E11-F030A	204A
2E11-F030B	204B
2E11-F030C	204C
2E11-F030D	204D
2E11-F041A	54C
2E11-F041B	32C
2E11-F041C	54A

MPL NUMBER	PEN. NUMBER
2E11-F041D	32A
2E11-F055A	224A
2E11-F055B	224B
2E11-F097	210A
2E11-F103A	224A
2E11-F103B	224B
2E11-F3078A	224A
2E11-F3078B	224B
2E21-F001A	208A
2E21-F001B	208B
2E21-F005A	16A
2E21-F005B	16B
2E21-F015A	226A
2E21-F015B	226B
2E21-F018A	79A
2E21-F018B	40D (F)
2E21-F018C	40A (C)
2E21-F036A	226A
2E21-F036B	226B
2E21-F044A	226A
2E21-F044B	226B
2E41-F002	11
2E41-F003	11
2E41-F012	210B
2E41-F021	214
2E41-F022	215
2E41-F024A	79F
2E41-F024B	40D (A)
2E41-F024C	79E
2E41-F024D	40D (B)
2E41-F040	215
2E41-F042	207
2E41-F046	210B
2E41-F049	214

TABLE T7.0-2 (Sheet 3 of 4)
PRIMARY CONTAINMENT ISOLATION DEVICES: MPL TO PENETRATION NUMBER CROSS REFERENCE

MPL NUMBER	PEN. NUMBER	MPL NUMBER	PEN. NUMBER	MPL NUMBER	PEN. NUMBER
2E41-F051	207	2G51-F002	218A	2T48-D006	205
2E41-F104	214	2G51-F011	234A	2T48-F103	25
2E41-F104	221B	2G51-F012	234A	2T48-F104	25
2E41-F111	214	2P21-F032	46	2T48-F104	205
2E41-F111	221B	2P21-F034	46	2T48-F113	81
2E41-F121	206C	2P33-F002	3	2T48-F114	81
2E41-F122	206C	2P33-F003	60A	2T48-F115	235B
2E51-F001	212	2P33-F004	28	2T48-F116	235B
2E51-F002	213	2P33-F005	64	2T48-F118A	25
2E51-F003	203	2P33-F006	217A	2T48-F118B	205
2E51-F007	10	2P33-F007	217B	2T48-F209	69
2E51-F008	10	2P33-F010	3	2T48-F210	69
2E51-F019	210A	2P33-F011	60A	2T48-F211	233
2E51-F021	210A	2P33-F012	28	2T48-F212	233
2E51-F028	213	2P33-F013	64	2T48-F307	25
2E51-F031	203	2P33-F014	217A	2T48-F308	25
2E51-F040	212	2P33-F015	217B	2T48-F309	205
2E51-F044A	34F	2P42-F051	23	2T48-F310	205
2E51-F044B	51B	2P42-F052	24	2T48-F311	205
2E51-F044C	34E	2P51-F513	21	2T48-F318	220
2E51-F044D	51A	2P51-F651	21	2T48-F319	26
2E51-F104	212	2P64-F045	47	2T48-F320	26
2E51-F104	221C	2P64-F047	48	2T48-F321	44
2E51-F105	212	2P70-F002	63	2T48-F322	44
2E51-F105	221C	2P70-F003	63	2T48-F323A Air Cyl.	225G
2G11-F003	19	2P70-F004	22	2T48-F323B Air Cyl.	225H
2G11-F004	19	2P70-F005	22	2T48-F323CAir Cyl.	225J
2G11-F019	18	2P70-F066	51C	2T48-F323D Air Cyl.	225K
2G11-F020	18	2P70-F067	51C	2T48-F323E Air Cyl.	225L
2G11-F852	55	2P70-N003	22	2T48-F323F Air Cyl.	225M
2G11-F853	55	2P70-N016	51C	2T48-F323G Air Cyl.	225A
2G31-F001	14	2T23-F004	34C	2T48-F323H Air Cyl.	225B
2G31-F004	14	2T23-F005	34C	2T48-F323I Air Cyl.	225C
2G51-D001	218A	2T48-D006	25	2T48-F323J Air Cyl	225D

TABLE T7.0-2 (Sheet 4 of 4)
PRIMARY CONTAINMENT ISOLATION DEVICES: MPL TO PENETRATION NUMBER CROSS REFERENCE

MPL NUMBER	PEN. NUMBER
2T48-F323K Air Cyl.	225E
2T48-F323L Air Cyl	225F
2T48-F324	205
2T48-F325	230
2T48-F326	220
2T48-F327	230
2T48-F328A	205
2T48-F328B	205
2T48-F332A	231
2T48-F332B	235A
2T48-F333A	231
2T48-F333B	235A
2T48-F334A	80
2T48-F334B	67
2T48-F335A	80
2T48-F335B	67
2T48-F338	220
2T48-F339	220
2T48-F340	26
2T48-F341	26
2T48-F342A	225G
2T48-F342B	225H
2T48-F342C	225J
2T48-F342D	225K
2T48-F342E	225L
2T48-F342F	225M
2T48-F342G	225A
2T48-F342H	225B
2T48-F342I	225C
2T48-F342J	225D
2T48-F342K	225E
2T48-F342L	225F
2T48-F361A	206C
2T48-F361B	206A

MPL NUMBER	PEN. NUMBER
2T48-F362A	206F
2T48-F362B	206H
2T48-F363A	34D
2T48-F363B	51D
2T48-F364A	220
2T48-F364B	205
2T49-F002A	61A
2T49-F002B	15
2T49-F004A	221A
2T49-F004B	222B
2T49-F009A	224A
2T49-F009B	224B

MPL NUMBER	PEN. NUMBER

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3.5 Control Room Log #3

Learning Objectives:

1. Determine if any Technical Specification action statements are in effect.

2. Determine if any systems addressed in the introduction are in an abnormal alignment.

3.5.1 Introduction

During this technical specification session the limiting conditions for operation, bases, and application of limiting conditions for operation are addressed for the Instrumentation and Administrative controls sections.

During your control room tour at about 0800 hours, with the plant at 93% power and in the run mode, you identify the following conditions:

- One RFP/MT high water level trip circuit inop
- LPRM 32-39A high alarm

During further review of the logs you find that APRM-B was bypassed at 0019, as a result of a LPRM that failed high and causing a half scram. The operators bypassed the APRM and logged the event in the degraded equipment log. In addition, you were ask to resolve a discussion between two operators concerning the requirement(s) to make a change to the bases of technical specifications.

From the above information and the aid of technical specification, address the learning objectives.

3.5.2 Reactor Protection System

The Reactor Protection System (RPS) initiates a reactor scram when one or more monitored parameters exceed their specified limits, to preserve fuel cladding integrity, reactor coolant system integrity, and minimize the energy which must be absorbed following a LOCA.

The protection and monitoring functions of the RPS have been designed to ensure safe operation of the reactor. This is achieved by specifying limiting safety system settings in terms of parameters directly monitored by the RPS, as well as LCOs on other reactor system parameters. The limiting safety system settings are defined as the allowable values, which, in conjunction with LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits, including safety limits during design basis accidents.

Functional diversity is provided by monitoring a wide range of ***dependent and independent*** parameters.

3.5.3 Average Power Range Monitor

The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux increases. The APRM channels receive input signals from the local power range monitors (LPRMs) within the reactor core to provide an indication of the power distribution and local power changes. The APRM channels average these LPRM signals to provide a continuous indication of average reactor power from a few percent to greater than RTP. The APRM System is divided into 4 APRM channels and 4 two-out-of-four voter channels. Each APRM channel provides inputs to each of the four voter channels. The four voter channels are divided into two groups of two each, with each group of two providing inputs to one RPS trip system. The APRM System is designed to allow one APRM channel, but no voter channels, to be bypassed. A trip from any one unbypassed APRM will result in a "half-trip" in all four voter channels, but no trip inputs to either RPS trip system. A trip from any two unbypassed APRM channels will result in a full-trip in each of the four voter channels, which in turn results in two trip inputs into each RPS trip logic channel (A1, A2, B1, and B2). Three of the four APRM channels and all four of the voter channels are required to be OPERABLE to ensure that no single failure will preclude a scram on a valid signal. In addition, to provide adequate coverage of the entire core, consistent with the design bases for APRM functions, at least 17 LPRM inputs, with at least three LPRM inputs from each of the four axial levels at which the LPRMs are located, are required for each APRM channel.

3.5.3.1 APRM Neutron Flux—High (Setdown)

For operation at low power (i.e., MODE 2), the Average Power Range Monitor Neutron Flux—High (Setdown) Function is capable of generating a trip signal that prevents fuel damage resulting from abnormal operating transients in this power range. For most operation at low power levels, the Average Power Range Monitor Neutron Flux—High (Setdown) Function will provide a secondary scram to the Intermediate Range Monitor Neutron Flux—High Function because of the relative setpoints. With the IRMs at Range 9 or 10, it is possible that the Average Power Range Monitor Neutron Flux—High (Setdown) Function will provide the primary trip signal for a corewide increase in power. No specific safety analyses take direct credit for the Average Power Range Monitor Neutron Flux—High (Setdown) function. However, this function indirectly ensures that before the reactor mode switch is placed in the run position, reactor power does not exceed 25% RTP (SL 2.1.1.1) when operating at low reactor pressure and low core flow. Therefore, it indirectly prevents fuel damage during significant reactivity increases with THERMAL POWER < 25% RTP.

The Allowable Value is based on preventing significant increases in power when THERMAL POWER is < 25% RTP.

The Average Power Range Monitor Neutron Flux—High (Setdown) Function must be OPERABLE during MODE 2 when control rods may be withdrawn since the potential for criticality exists.

In MODE 1, the Average Power Range Monitor Neutron Flux—High Function provides protection against reactivity transients and the RWM and rod block monitor protect against control rod withdrawal error events.

3.5.3.2 APRM Flow Biased Simulated Thermal Power High

The Average Power Range Monitor Simulated Thermal Power—High Function monitors neutron flux to approximate the THERMAL POWER being transferred to the reactor coolant. The APRM neutron flux is electronically filtered with a time constant representative of the fuel heat transfer dynamics to generate a signal proportional to the THERMAL POWER in the reactor. Changes to fuel design include an evaluation of the time constant to determine if the electronic filter requires replacement. The trip level is varied as a function of recirculation drive flow (i.e., at lower core flows, the setpoint is reduced proportional to the reduction in power experienced as core flow is reduced with a fixed APPLICABILITY control rod pattern) but is clamped at an upper limit that is always lower than the Average Power Range Monitor Neutron Flux—High Function Allowable Value.

The Average Power Range Monitor Simulated Thermal Power—High Function provides protection against transients where THERMAL POWER increases slowly (such as the loss of feedwater heating event) and protects the fuel cladding integrity by ensuring that the MCPR SL is not exceeded. During these events, the THERMAL POWER increase does not significantly lag the neutron flux response and, because of a lower trip setpoint, will initiate a scram before the high neutron flux scram. For rapid neutron flux increase events, the THERMAL POWER lags the neutron flux and the Average Power Range Monitor Neutron Flux—High Function will provide a scram signal before the Average Power Range Monitor Simulated Thermal Power—High Function setpoint and associated time delay are exceeded. Each APRM channel uses one total drive flow signal representative of total core flow. The total drive flow signal is generated by the flow processing logic, which is part of the APRM channel. The flow is calculated by summing two flow transmitter signals, one from each of the two recirculation loop flows. The flow processing logic OPERABILITY is part of the APRM channel OPERABILITY requirements for this Function.

The clamped Allowable Value is based on analyses that take credit for the Average Power Range Monitor Simulated Thermal Power—High Function for the mitigation of the loss of feedwater heating event. The time constant is based on the fuel heat transfer dynamics and provides a signal proportional to the THERMAL POWER.

The Average Power Range Monitor Simulated Thermal Power—High Function is required to be OPERABLE in MODE 1 when there is the possibility of generating excessive THERMAL POWER and potentially exceeding the SL applicable to high pressure and core flow conditions (MCPR SL). During MODES 2 and 5, other IRM and APRM Functions provide protection for fuel cladding integrity.

3.5.3.3 APRM Fixed Neutron Flux High

The Average Power Range Monitor Neutron Flux—High Function is capable of generating a trip signal to prevent fuel damage or excessive RCS pressure. For the overpressurization protection analysis of Reference 4, the Average Power Range Monitor Neutron Flux—High Function is assumed to terminate the main steam isolation valve (MSIV) closure event and, along with the safety/relief valves (S/RVs), limits the peak reactor pressure vessel (RPV) pressure to less than the ASME Code limits. The control rod drop accident (CRDA) analysis (Ref. 7) takes credit for the Average Power Range Monitor Neutron Flux—High Function to terminate the CRDA.The Allowable Value is based on the Analytical Limit assumed in the CRDA analyses.

The Average Power Range Monitor Neutron Flux—High Function is required to be OPERABLE in MODE 1 where the potential consequences of the analyzed transients could result in the SLs (e.g., MCPR and RCS pressure) being exceeded. Although the Average Power Range Monitor Neutron Flux—High Function is assumed in the CRDA analysis, which is applicable in MODE 2, the Average Power Range Monitor Neutron Flux—High (Setdown) Function conservatively bounds the assumed trip and, together with the assumed IRM trips, provides adequate protection. Therefore, the Average Power Range Monitor Neutron Flux—High Function is not required in MODE 2.

3.5.3.4 APRM Inop

This Function (Inop) provides assurance that the minimum number of APRM channels is OPERABLE. For any APRM channel, any time:

1) its mode switch is in any position other than "Operate," 2) an APRM module is unplugged, or 3) the automatic self-test system detects a critical fault with the APRM channel, an Inop trip signal is sent to all four voter channels. Inop trips from two or more unbypassed APRM channels result in a trip output from all four voter channels to their associated trip system.

This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis. There is no Allowable Value for this function. This function is required to be OPERABLE in the MODES where the APRM Functions are required.

3.5.3.5 Two-out-of-Four Voter

The Two-out-of-Four Voter Function provides the interface between the APRM Functions and the final RPS trip system logic. As such, it is required to be OPERABLE in the MODES where the APRM Functions are required and is necessary to support the safety analysis applicable to each of those Functions. Therefore, the Two-out-of-Four Voter Function is required to be OPERABLE in MODES 1 and 2.

All four voter channels are required to be OPERABLE. Each voter channel also includes self-diagnostic functions. If any voter channel detects a critical fault in its own processing, an Inop trip is issued from that voter channel to the associated trip system.

There is no Allowable Value for this Function.

3.5.4 Feedwater and Main Turbine High Water Level Instrumentation

The feedwater and main turbine high water level trip instrumentation is assumed to be capable of providing a turbine trip in the design basis transient analysis for feedwater controller failure, maximum demand event. The high level trip indirectly initiates a reactor scram from the main turbine trip (above 30% power) and trips the feedwater pumps, thereby terminating the event. The reactor scram mitigates the reduction in MCPR.

3.5.5 Administrative Controls

The General Manager - Nuclear Plant shall provide direct executive oversight over all aspects of the plant. The Assistant General Manager-Plant Operations shall be responsible for overall unit operation, and delegates, in writing the succession of this responsibility. A staff of Shift Supervisors, each licensed as Senior Reactor Operator (SRO), reports to the Assistant General Manager-Plant Operations and carries on-shift management responsibilities for safe operation of the plant. The Operating Supervisor is the SRO in charge of reactor operations on shift. Normally the Operating Supervisor stands watch in the control room, however, he/she may leave when the Shift Supervisor is present in the control room. The Shift Supervisor is responsible for all site activities in the absence of the Plant manager or designated alternates.

3.5.5.1 Procedures

Written procedures shall be established, implemented and maintained covering the activities referenced below:

- The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978.
- Refueling operations.

- Surveillance and test activities of safety related equipment.
- Security Plan implementation
- Emergency Plan implementation
- Fire Protection Program implementation
- Process Control Program implementation
- ODCM implementation

Appendix A, Regulatory Guide 1.33

Appendix A, of Regulatory Guide 1.33, list typical safety related activities that should be covered by written procedures. This appendix is not intended as an inclusive listing of all needed procedures since many other activities carried out during the operation phase of nuclear power plants should be covered by procedures not included in this list.

3.5.5.2 Reportable Event Action

The following actions shall be taken for reportable events:

- The commission shall be notified and/or a report submitted pursuant to requirements of section 50.73 to 10 CFR part 50, and
- Each reportable event shall be reviewed by the PRB, and the results of this review shall be submitted to the SRB, the General Manager - Nuclear Plant, and the Vice President - Nuclear.

3.5.5.3 Safety Limit Violation

The following actions shall be taken in the event a safety limit is violated:

- Within 1 hour, notify the NRC Operation Center, in accordance with 10 CFR 50.72.
- Within 2 hours:
  Restore compliance with All SLs; and
  Insert all insertable control rods.
- Within 24 hours, notify the plant manager, the corporte executive responsible for overall plant safety, and the offsite review committee.
- Within 30 days, a LER shall be prepared pursuant to 10 CFR 50.73. The LER shall be summitted to the NRC, the offsite review committee, the plant manager, and corporate executive responsible for overall plant nuclear safety.
- Opeartion of the unit shall not be resumed until authorized by the NRC.

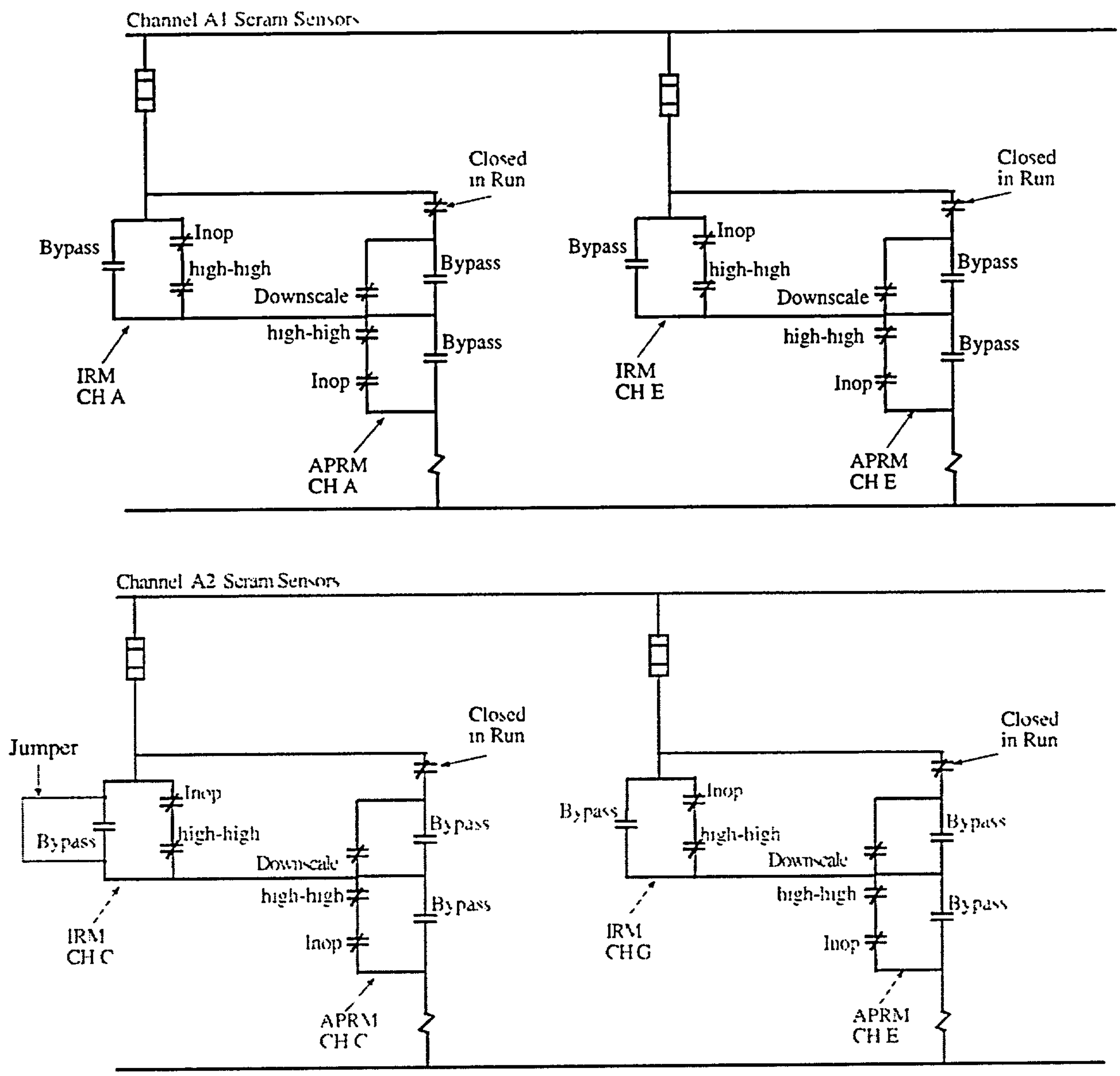
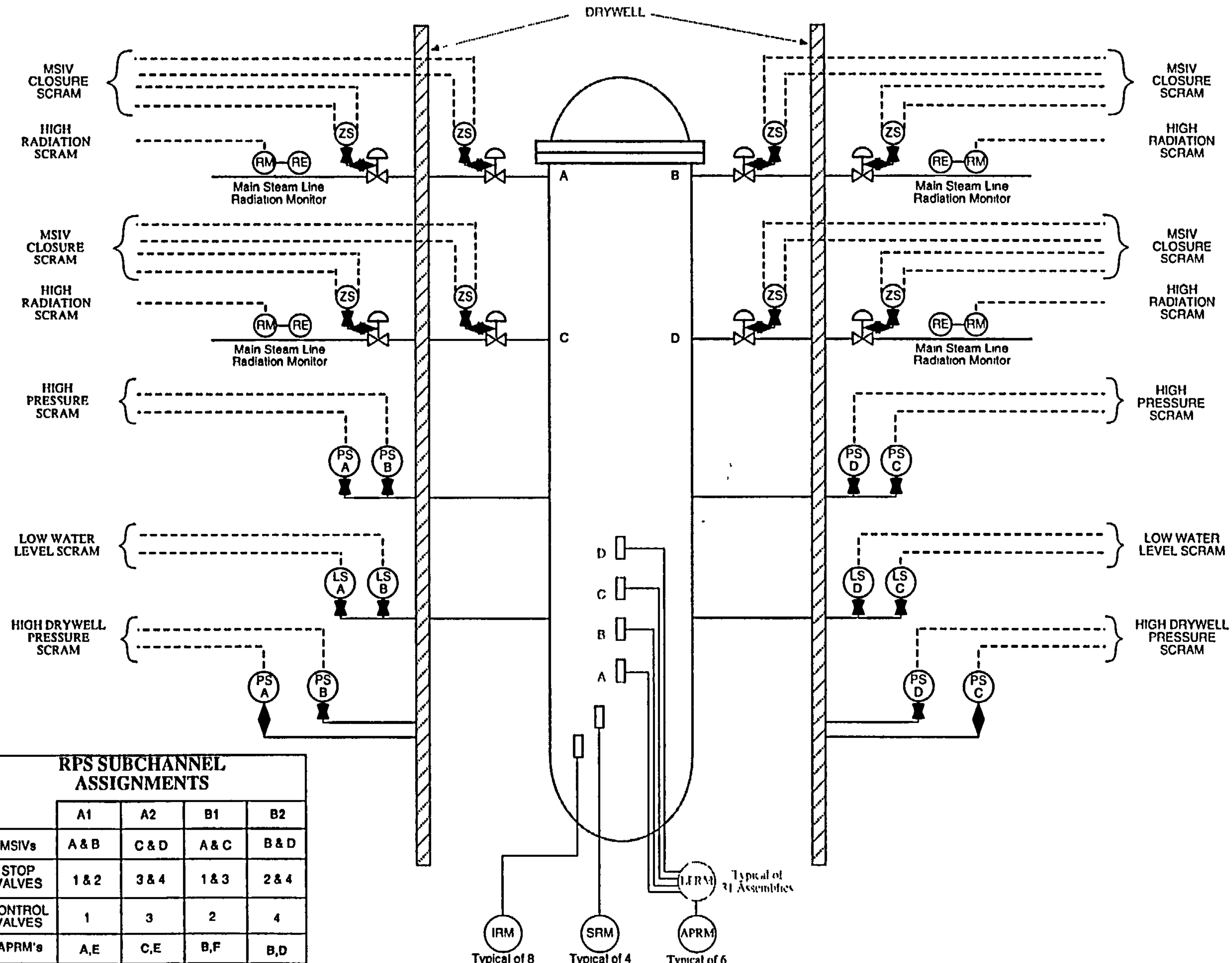


Figure 3.5-1 IRM/APRM - RPS Interface



RPS SUBCHANNEL ASSIGNMENTS

	A1	A2	B1	B2
MSIVs	A & B	C & D	A & C	B & D
STOP VALVES	1 & 2	3 & 4	1 & 3	2 & 4
CONTROL VALVES	1	3	2	4
APRM's	A,E	C,E	B,F	B,D
IRM's	A E	C,G	B,F	D,H

Figure 3.5-2 Reactor Protection System Sensors

Boiling Water Reactor
GE BWR/4
Technology Advanced Manual

Chapter 4.0

Technical Issues

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4.1 BWR ECCS EVOLUTION

4.1.1 Loss of Coolant Accident (LOCA)

The most severe accident (design basis accident) used for purposes of containment design, is the steam line break. Analyzing breaks in the main steam line covers the effects of all other steam type breaks.

The design basis accident used for the purposes of establishing core performance and cladding integrity is the instantaneous "guillotine" rupture of a recirculation line. Depending on plant design, a break in the suction or discharge line may be the worst case. Break size and location determines how fast pressure will decrease to allow the low pressure injection system to reflood the core. Spectrum analysis performed on a recirculation line break covers the effect of all other type liquid breaks such as the RHR suction and return lines, and recirculation riser lines.

For a given size break, the lower the elevation at which the broken line penetrates the reactor vessel, the greater will be the resultant peak clad temperature (PCT); i.e., PCT will be higher for those lines penetrating the vessel area that contain water than those penetrating the vessel steam space. Thus, to demonstrate the performance and capability of the Emergency Core Cooling System (ECCS), recirculation line breaks were analyzed since those resulted in the highest peak clad temperatures for a given break size.

4.1.2 Pre-LOCA Initial Conditions

In order to calculate the amount of potential fuel damage that could occur during and following a LOCA, a set of initial fuel and core conditions are specified. The values are conservatively chosen to be greater than those expected during normal full power operations. The resultant calculations are used to establish ECCS acceptance criteria. ECCS equipment is subsequently evaluated to determine if individually or collectively they are capable of preventing the plant from exceeding those criteria. Initial conditions include the following:

102% Reactor Power -

A calculated amount of stored heat that is possible to obtain due to the reactor power being at this level for an indefinite period of time. Even though this is unlikely to occur, this power level is used so as to include margins for instrument error.

600°F Cladding Temperature -

At the time of the LOCA the cladding would be at a temperature near that of the adjacent coolant or approximately 600°F.

2000°F UO_2 Average Temperature and 4000°F Peak Centerline Temperature -

The average temperature and peak centerline temperature are selected as calculated temperatures at the onset of the LOCA. Its realized that the hottest fuel pellets (hotspots) will be well above both of these values.

The excess heat that is contained in the fuel pellets is called stored heat and is approximately proportional to the power density and the thermal resistance of the pellet to clad gap. Stored heat is an important factor because it will significantly contribute to the cladding temperatures during the LOCA scenario.

4.1.3 LOCA Event Sequence

In order to emphasize the potential consequences of a LOCA, the expected sequence of events must be clearly understood. Although the DBA has not occurred at an operating commercial nuclear power plant, the core thermal and hydraulic responses to such an event is well known and predictable. Test facility experimentation was performed during the late 1950's and early 1960's which included the Boiling Water Reactor experiments (BORAX), Experimental Boiling Water Reactor (EBWR), and

Special Power Excursion Reactor Test (SPERT).

Also, computer analysis and statistical models including the General Electric Thermal Analysis Basis (GETAB) and the General Electric Critical Quality (X)/Boiling Length (L),(GEXL correlation), provide conservative calculations of critical power and the occurrence of boiling transition within a fuel channel.

The early tests and experiments, and intricate computer codes, analysis, and models, provides credence to the expected reactor response to a LOCA. The sequence of events for such an accident is described as follows:

- Fission heat drops rapidly - This occurs due to rapid void formation and the reactor scram.
- Clad cooling decreases - Core flow decreases suddenly when recirculation pumps trip. Also, moderator density is reduced as the bulk coolant flashes to steam and blankets the outside of the clad surface.
- Pellet temperatures equalize - Fuel pellet centerline temperature initially decreases as fission heat production drops and stored heat is removed by the steam-water mixture produced during the blowdown phase.
- Pellet temperatures begin increasing - Decay heat provides a continued source of heat which can no longer be removed when vessel blowdown is completed.
- Zircaloy oxidation - If the cladding temperature exceeds 1800°F, oxidation will occur. The chemical process adds additional heat to the cladding and also causes pellet temperatures to increase.
- Pellet temperature increases until reflood begins - As vessel reflood commences, much of the water flashes to steam. It is the steam in cobination with entrained water droplets that provides initial cooling of the core. As reflood continues, sufficient cooling is provided to overcome the heat inputs from the decay of fission products and cladding oxidation.
- Clad heatup is terminated - Continued injection of coolant by the ECCS will eventually cover the core with water.

4.1.4 Cladding Failure Mechanisms

In order to maintain the integrity of the fuel rods, cladding ductility must be maintained. Metallurgical and chemical changes will effect ductility.

Zirconium has two different metallurgical crystal structures including the alpha phase and the beta phase. At room temperatures zirconium is in the alpha phase which is a brittle crystal structure. When heated above 1150°F, the crystal structure undergoes a change and is transformed into the beta phase which is ductile. However, if the zirconium cladding oxidizes, even though its temperature is above 1150°F, the crystal structure is in the alpha phase and becomes brittle.

Oxidation of the cladding is a chemical event that occurs due to a steam oxidation process and is normally referred to as a metal-water reaction. Water molecules are absorbed on the surface of the cladding and disassociate to hydrogen and hydroxyl radicals at high temperatures. Within the surface of the cladding the hydroxyl radicals, after several chemical steps, are converted into oxygen ions and hydrogen atoms. The hydrogen atoms, wherever formed, will combine into hydrogen molecules and escape from the surface of the cladding. The oxygen ions however, diffuse further into the surface and are dissolved into the metal. As this reaction continues and if the concentration of oxygen is high enough, zirconium dioxide is formed. This oxidation process takes place between 2060 and 2960°F. The formation of zirconium dioxide causes this area of the cladding to become brittle and the loss of ductility of this metal may cause the fuel rods to burst upon quenching. The thickness and the rate of oxidation is temperature dependent.

4.1.5 ECCS Criteria Development

4.1.5.1 Initial ECCS Criteria

The original standard for the BWR ECCS was developed as a result of lengthy discussions and agreements between the General Electric Company (GE), Westinghouse (W) and the Atomic Energy Commission (AEC). That standard established the first ECCS criteria listed below:

- Peak cladding temperature - 2700°F
- Two independent and separate physical means for ensuring adequate core cooling following a LOCA (e.g. spray and flooding)

4.1.5.2 Interim ECCS Criteria

On June 29, 1971, the AEC published the interim criteria for immediate implementation. The vendors were allowed a reasonable amount of time to compile data and comments concerning those criteria prior to commencing public hearings which would establish the final ECCS acceptance criteria. Those interim criteria provided the basis for reasonable assurance that the ECCS would effectively limit core damage in the highly unlikely event of a LOCA. The interim criteria consisted of the following:

- Peak Cladding Temperature - 2300°F
- Maximum hydrogen generation - The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed .01 times the hypothetical amount that would be generated if all the cladding were to react.
- Coolable geometry - The calculated changes in core geometry shall be such that the core remains amenable to cooling.
- Long Term Cooling - After any calculated initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and the decay heat shall be removed for an extended period of time.

Additionally, single failures resulting in complete loss of the Low Pressure Coolant Injection (LPCI) System was required to be considered in the accident sequence and analyses.

4.1.5.3 Final ECCS Acceptance Criteria

Public hearings commenced early in 1972 for the purpose of assisting the AEC in its determination as to whether or not the Interim Criteria should be retained as issued or if those criteria should be adopted in some other form. Participation in the hearings was extensive. In attendance were members of the AEC, three states, four reactor vendors, a consolidated group of electric utility companies, intervenors, and other interested parties and individuals. The hearings lasted a total of 125 days and generated more than 22,000 pages of transcript.

After all documentation was submitted and testimonies heard, the AEC made its decisions and the final acceptance criteria was published December 28, 1973. Some of the criteria were highly contested by the vendors and utility groups. Those arguments and the basis for the criteria will be discussed later. In 1974, the final criteria was added as paragraph 50.46 of 10 CFR 50.

Interim criterion number one, specifying that the temperature of the Zircaloy cladding should not exceed 2300°F, was replaced by the first two criteria listed below. The other three criteria were retained with some modification of the wording. Single failure consideration continued to be required. The final criterion and the basis of each are as follows:

Peak Cladding Temperature - 2200°F Basis

This criteria and the one which follows (17% clad oxidation) are closely interrelated and together ensure that the zircaloy cladding will remain sufficiently intact to retain the UO_2 fuel pellets in their separate fuel rods and therefore remain in an easily coolable array. Conservative calculations indicate that the cladding will swell and burst in a longitudinal split but will retain the fuel pellets provided that the cladding is not too heavily oxidized.

Maximum Cladding Oxidation - 17% of the total cladding thickness before oxidation.

Same as Basis given above.

Maximum Hydrogen Generation - 1% of the hypothetical amount that would be generated if all the cladding were to undergo a zirconium-water reaction. Basis

This criteria ensures that hydrogen will not be generated in amounts that could lead to explosive concentrations. This criteria is the same as the interim acceptance criteria with the exception that it is more explicit in detailing how much of the Zircaloy is to be used as the bases for the 1% hydrogen calculation.

Coolable Geometry Basis

Calculated changes in core geometry shall be such that the core remains amenable to cooling.

Long Term Cooling Basis

After any calculated initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and the decay heat shall be removed for an extended period of time.

The long term maintenance of cooling is considered from the time the cladding is cooled to 300°F or less. The intent of this criteria is self-evident.

4.1.5.4 Appendix K

In 1974, a new Appendix K titled "Appendix K- ECCS Evaluation Models," was added to 10 CFR Part 50. Appendix K provides guidelines for the ECCS evaluation models. Some highlights of this appendix include:

- Gives the terms and assumed values that are to be used in the ECCS design evaluations. When computer calculations are performed to evaluate the effectiveness of the ECCS and maximum temperatures in the core, the following conservative values are used:
  - Stored heat - assumes that the reactor was at 102% for an indefinite period of time with the highest peaking factors.
  - Blow down - heat transfer during blowdown is made in the conservative direction. It is probable that this factor has a conservatism of several hundred degrees fahrenheit.
  - Heat generation rate - it is assumed that the heat generation rate from the decay of fission products is 20% greater than proposed by ANS standards. This figure is used due to

operating at 102% of rated power for an infinite time which represents an improbable situation.

- Peak temperature criteria - limitation of the peak calculated temperature of the cladding (2200°F) is applied to the hottest region of the hottest fuel rods. This provides a substantial degree of conservatism to ensure that the core will suffer a very limited amount of core damage due to a LOCA.

- ECCS single failure criteria. These calculations also have to assume the most damaging single failure of the ECCS component or subsystem.

- Addresses reflood and refill rates of less than 1 inch per second i.e., if the reflood/refill rate drops to less than 1 inch per second, then the calculations must assume that the cooling of the core is by steam alone. This is very conservative because the water splatter carryover that will be entrained in the steam will remove heat from the cladding but is not used in the calculations.

Many years have passed since Appendix K was implemented. Calculations have been revised as a direct result of obtaining better data through research and development programs performed by NRC and private industry. Recent calculations have established that the maximum fuel clad temperatures reached during a LOCA will be approximately 900°F less than the older calculated value of 2200°F. This added margin of safety has resulted in the reduction of many restrictions in the areas of fuel operating temperatures, surveillance testing frequencies, and permissible "down times" for safety-related equipment for testing and maintenance. These changes should result in increased reactor availability and more efficient fuel burnup.

4.1.6 Meeting Changing ECCS Criteria

The initial criteria was met by an ECCS consisting of two 100% core spray (C.S.) systems and one low pressure coolant injection (LPCI) System. Original data and calculations proved that the C.S. System could by itself, terminate post accident heatup by spray action alone. Core spray or LPCI could successfully meet the peak cladding temperatures (PCT) for all large line breaks.

When the interim criteria was established, initial calculations indicated that PCT could not be maintained less than 2300°F. Several factors contributing to the inadequacy of the ECCS included:

- Establishment of the single failure criteria - the single worst failure was determined to be a failure within the LPCI system. The LPCI system included a LPCI loop selection logic that prevented opening of the injection valve supplying the "broken" recirculation loop and opened the injection valve which supplied the "good" loop thus supplying water from both divisions of LPCI. Failure of the injection valve supplying the "good" recirculation loop to open would render inoperable the entire LPCI System.

- The C.S. System was judged incapable of meeting the new PCT requirements by itself. This was due to the existing 7X7 fuel design and C.S. system test results that indicated counter current flow limiting effects and questionable spray behavior in a steam environment.

The reactor vendor and owners groups established several possible methods for meeting the interim criteria. Possible alternatives included:

- Redesign the LPCI and C.S. Systems in order to take credit for both spray and flooding.

- Redesign the fuel to limit power production by the fuel pellets.
- Take credit for water accumulation and the eventual flooding capability of the C.S. System.
- Limit PCT by limiting MAPLHGR.

A reanalysis was performed using the last two alternatives listed above and the results indicated that either the C.S. or LPCI systems could prevent PCT from exceeding 2300°F for all large pipe breaks. However, with LPCI unavailable, the C.S. System would be required to provide both spray and flooding. The flooding capability was accomplished by drilling holes in the lower core plates.

In core instrument tube vibrations on BWR/4 plants required plugging of the bypass flow holes in the lower core plate. Those holes provided part of the design core bypass flow (10%). The bypass holes also allowed the core spray water to accumulate in the bypass region to reflood the bottom head volume and then the fuel. Plugging the holes resulted in a reduction in the core sprays ability to reflood the core and maintain PCT below the specified limit. This new problem meant that on high power density cores the core spray system could not meet the final 2200°F criteria without severe MAPLHGR restrictions. This prompted General Electric to restore the bypass flow by drilling holes in the lower tie plate of the fuel assemblies.

The final acceptance criteria further restricted the maximum PCT to 2200°F. This limit made discharge pipe breaks more severe for certain vessel geometries. Also, because the LPCI System was assumed to be unavailable for those breaks, it was determined that the C.S. System may not prevent exceeding the PCT criteria in high power density cores even with combined spray and flooding capability.

Rather than placing further limits on MAPLHGR, the final acceptance criteria was met by a combination of design and physical plant changes. Those changes involved substantial changes to the LPCI System including removal of the LPCI Loop Selection Logic, permanent closure or removal of the LPCI pumps discharge lines Division I and II crossconnect valve, and total separation and independency of the two divisions.

Completing the modifications to the LPCI System provided assurance that the C.S. System in combination with all or one of the two divisions of LPCI would meet all of the new ECCS criteria for all size pipe breaks. Later development of the 8 X 8 fuel assemblies reduced MAPLHGR, thus also contributing to the margin of safety during the LOCA.

4.1.7 Vendors Response To The Final Acceptance Criteria

None of the reactor manufacturers or owners groups agreed with the Staffs proposal of 2200°F maximum cladding temperature. Westinghouse proposed a calculated temperature of 2700°F. Combustion Engineering and the utility group agreed on a calculated temperature of 2500°F because much of the data on oxidation and its effects stops at less than 2500°F. B&W suggested a more conservative figure of 2400°F because excessive metal- water reaction rates would be precluded below 2400°F. GE disagreed with the staffs position of 2200°F and stated that 2700°F was acceptable as far as embrittlement of the cladding was concerned and suggested that the interim criteria of 2300°F be retained to ensure that the core never gets into regions of metal-water reactions. Since the owners groups did not agree with each other and all had different calculated temperatures, the staff chose to retain its proposed value of 2200°F.

All owner groups essentially agreed on the oxidation limit of 17%. However, they had recommended that a more conservative limit of 12% be used to avoid brittle behavior of the cladding. It should be understood that the 12% value was to be used with a calculated higher cladding temperature than that proposed by the AEC staff.

The maximum hydrogen generation criteria was misunderstood by some of the vendors and their analysis. This criteria has nothing to do with oxidation or the need to retain the strength of the cladding.

Combustion Engineering and Babcock & Wilcox argued that a coolable geometry criteria was no longer necessary because the peak cladding temperature and oxidation criterion would ensure adequate fuel rod integrity. The AEC agreed with the vendors. However, the Commission felt that in view of the fundamental and historical importance of maintaining core coolability, they desired to retain this criteria as a basic objective.

The long term cooling criteria is the same as the interim criteria. No arguments were presented by the vendors in this area.

4.1.8 Summary

The Commission believes the implementation of the new regulations will ensure an adequate margin of performance of the ECCS should a design LOCA ever occur. This margin is provided by conservative features of the evaluation models and by the criteria themselves.

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4.2 ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS)

Learning Objectives :

1. Define anticipated transient without scram.
2. Explain the expected plant response for the worst ATWS event.
3. List the scram signals received during the initial ATWS event.
4. Explain why an ATWS event is safety significant.
5. List various ways to limit core power during an ATWS event.

4.2.1 Introduction

In general, the term reactor transient applies to any significant deviation from the normal operating value of any of the key reactor operating parameters. Transients may occur as a consequence of an operator error or the malfunction or failure of equipment.

Anticipated transients are deviations from the normal operating conditions that may occur one or more times during the service life of a plant. Anticipated transients range from trivial to significant in terms of the demands imposed on plant equipment. Anticipated transients include such events as a turbine trip, EHC failure, MSIV closure, loss of feedwater flow and loss of feedwater heating. More specifically, all situations (except for LOCA) which could lead to fuel heat imbalances are anticipated transients.

Many transients are handled by the reactor control systems, which would return the reactor to its normal operating conditions. Others are beyond the capability of the reactor control systems and require reactor shutdown by the Reactor Protection System (RPS) in order to avoid damage to the reactor fuel or coolant systems. If such a transient should occur and if, in spite of all the reliability built into the Reactor Protection System, a scram should not result, then an ATWS event would have occurred.

4.2.2 History

ATWS became a possible source of concern in nuclear power plants in 1968 during discussions between ACRS, the regulatory staff, and reactor instrument designers about the safety implications of interactions between normal control system circuitry and protection system circuitry in the instrument systems of power plants. After considerable discussion and some design changes, it was determined that separation of control and protection functions was being achieved to a reasonable degree, either by physical separation or by electrical isolation. The focus of interest with regard to instrument systems then shifted to the ability of the shutdown system to function with the needed reliability considering common mode failures. Common mode failures have to do with design or maintenance errors that might be made for similar redundant portions of a protection system. One of the difficult aspects of deciding whether or not common mode failures were being adequately accounted for in shutdown system design was that techniques to analyze a system for common mode failures were not as well developed as techniques to analyze a system for random failures.

4.2.2.1 1969

The efforts to evaluate the safety concerns of ATWS events went in two general directions. The first was concerned with attempting to evaluate the likelihood of common mode or other failures of the reactor protection system that could lead to ATWS events. The second was to assume, simply as a basis for discussion, that ATWS was possible and to examine the consequences of various postulated ATWS events.

4.2.2.2 1970

After analyzing vender supplied information it was concluded that several anticipated transients in BWRs would require prompt action to shutdown the reactor in order to avoid serious plant damage and possible offsite release. The resulting list of transients considered for boiling water reactor plants is as follows:

- **Primary Pressure Increase**

These transients include loss of load events such as generator trip, turbine trip, and loss of condenser vacuum. Also considered are such transients as closure of one or all of the main steam line isolation valves and malfunction of the reactor primary system pressure regulator, causing increasing pressure.

- **Reactor Water Inventory Decrease**

These transients include events leading to a decrease in the inventory of reactor primary coolant such as loss of auxiliary power, loss of feedwater, pressure regulator failure in a direction to cause decreasing reactor system pressure, inadvertent opening of a safety or relief valve, and opening of condenser bypass valves.

- **Reactor Coolant Flow Increase**

These transients include events that might increase the recirculation flow and thus induce a positive reactivity increment. They include a malfunction of the recirculation flow controller in a manner to cause increasing primary coolant flow and the start-up of a recirculation pump that had been on standby.

- **Reactor Water Temperature Decrease**

These transients include events that might cause a power surge by reduction of the reactor primary coolant water temperature. They include malfunction of feedwater control in a direction to increase feedwater flow, loss of a feedwater heater, shutdown cooling malfunction, and inadvertent activation of auxiliary cold water systems.

- **Reactivity Insertions**

These transients include control rod withdrawal transients from the zero reactor power, hot, critical condition and from full power; fuel assembly insertion; control rod removal; and control curtain removal errors during refueling.

- **Reactor Coolant Flow Decrease**

These transients include failure of one or more recirculation pumps or malfunction of the recirculation flow control in a direction to cause decreasing flow.

The transients having the greatest potential for significant damage are those leading to a reactor primary coolant system pressure increase. The most severe of these are the loss of condenser vacuum and the closure of all main steam isolation valves. A loss of condenser vacuum causes automatic closure of the turbine stop valves and the turbine bypass valves. The turbine stop valves are fast acting valves, so that there is an abrupt interruption of steam flow from the reactor. The main steam isolation valves are slower in closing, but in this case the large steam line volume is not available to buffer the pressure rise. The result in either case would be an increase in primary system pressure and temperature. The pressure increase would decrease the volume of steam bubbles in the reactor core and this, in turn, would increase the reactivity and cause a surge in reactor power. The power surge would cause a further increase in system temperature and pressure, with the pressure rising to values above acceptable limits. The other transients that lead to primary system pressure increase are less severe. Generator or turbine trips are less severe because the turbine bypass valves can be assumed to open and the condenser to be operative. Although the transient proceeds more slowly in these cases, the result still would be an excessively high reactor coolant system pressure.

4.2.2.3 1971

The ACRS and the regulatory staff concluded that a design change to the proposed Newbold Island (now Hope Creek) BWR/4 (Public Service of New Jersey) was appropriate to limit the possible consequences of ATWS. The same design change was applied to other BWR/4s. The design change consisted of tripping of the recirculation pumps.

4.2.2.4 1972

The ACRS recognizes ATWS as a low probability event. Nevertheless, it believed that, in consideration of the large number of BWRs expected eventually to be in operation, and in view of the expected occurrence rate of anticipated transients, experience with scram systems of current design is insufficient to give assurance of an adequately low probability for an ATWS event with possible serious consequences. Accordingly a set of positions and actions is implemented and was published as WASH-1270.

4.2.2.5 1973

The regulatory staff amends licensing position setting October 1, 1973 as the effective date of the position. Analyses for older operating plants should be provided by October 1, 1974, and the need for any changes would be considered on a case-by-case basis. Plants recently started in operation, now under construction, or for which applications for construction permits are filed before October 1, 1976, should have any equipment provided and any changes made that are necessary to make the consequences of ATWS acceptable. Analyses of the effects of ATWS and plans and schedules for any changes found necessary should be provided for these plants by October 1, 1974, or at the time of submission of an application for a construction permit, whichever is later. Plants for which applications for construction permits are filed after October 1, 1976, should have improvements in the protection system design that make an ATWS event negligibly small.

Applicants should be required to:

- demonstrate that with their present designs the consequences of anticipated transients without scram (ATWS) are acceptable,
- or make design changes which render the consequences of anticipated transients without scram acceptable,
- or make design changes to improve significantly the reliability of the scram system.

It is necessary to establish acceptable consequences of ATWS in order to implement either option 1 or option 2 of the recommended position. Acceptable conditions are defined as follows:

- **Radiological Consequences**

The radiological consequences shall be within the guideline values set forth in 10 CFR Part 100.

- **Primary System Pressure**

The maximum acceptable transient primary system pressure shall be based on the primary system pressure boundary limit or the fuel element limit, whichever is more restrictive. Primary pressure boundary limits transient pressure shall be limited to less than that resulting in a maximum stress anywhere in the reactor coolant pressure boundary of the "emergency conditions" as defined in the ASME Section II Nuclear Power Plant Components Code.

Fuel pressure limits transient pressure shall not exceed a value for which test and/or analysis demonstrate that there is no substantial safety problem with the fuel.

- **Fuel Thermal and Hydraulic Effects**

The increase in fuel enthalpy shall not result in significant cladding degradation or in significant melting of fuel even in the hottest fuel zones.

- **Containment Conditions**

Calculated containment pressures shall not exceed the design pressure of the containment structure. Equipment which is located within the containment and which is relied upon to mitigate the consequences of ATWS shall be qualified by testing in the combined pressure, temperature and humidity environment conservatively predicted to occur during the course of the event.

- **Analyses of Possible Detrimental Effects of Required Modifications**

Any modifications made to comply with option 2 of the recommended position shall be shown not to result in violations of safety criteria for steady state, transient, or accident conditions and shall not substantially affect the operation of safety related systems.

- **Diversity Requirement for Implementing Option 2 of the Recommended Position**

Design changes to make the consequences of ATWS acceptable should not rely on equipment or system designs which have a failure mode common with the scram system. The equipment involved in the design change shall, to the extent practical, operate on a different principle from equipment in the scram system. As an absolute minimum, the equipment relied on to render acceptable the consequences of the ATWS event shall not include equipment identical to equipment in the associated scram system.

- **Diversity Requirement for Implementing Option 3 of the Recommended Position**

Improvements must reduce considerably the potential for common mode failure of the scram system. Failures of identical equipment from a common mode should not disable sensing circuits, logic, actuator circuits or control rods to the extent that scram is ineffective. The addition of a separate protection system utilizing principles diverse from the primary protection system is indicated in order to meet this requirement.

4.2.2.6 1974

Reactor vendors submitted analyses on ATWS in general response to the following requirements set forth in WASH-1270:

- Trip of the reactor recirculation pump upon high reactor vessel pressure or low low reactor water level.
- Logic for automatic initiation of the liquid control system.
- Add piping to supply some of the liquid control flow through the HPCI system.

4.2.2.7 1975

ATWS is almost resolved. With the issuance of WASH-1400 (Assessment of Accident Risks), reactor vendors turned to the results which demonstrated that ATWS was not a major contributor to the risk from LWRs and as such no modifications are required.

4.2.2.8 1976

ATWS remained a controversial issue between the NRC and the industry.

4.2.2.9 1977

NRC formed a task force on ATWS in an effort to finally resolve the matter. The report sent to ACRS, reiterated the general position of scram unreliability which could not be shown to be acceptable low and measures were required to mitigate the consequences of ATWS. The year 1977 passed without issuance of a new NRC position on ATWS.

4.2.2.10 1978

The NRC issues NUREG-0460 (Anticipated Transients Without Scram for Light Water Reactors). The NUREG includes the Following:

- **ATWS Acceptance Criteria**

The staff recommends that all nuclear power plant designs should incorporate the designs features necessary to assure that the consequences of ATWSs would be acceptable. The primary criterion for acceptability is that the calculated radiological consequences must be within the dose guidelines values set forth in 10 CFR Part 100. In addition, more specific acceptance criteria have been developed for primary system integrity, fuel integrity, containment integrity, long-term shutdown and cooling capability, and the design of mitigating systems.

- **Containment Integrity**

The calculated containment pressure, temperature and other variables shall not exceed the design values of the containment structure, components and contained equipment, systems or components necessary for safe shutdown. For boiling water reactor pressure suppression containments, the region of relief or safety valve discharge line flow rates and suppression pool water temperatures where steam quenching instability could result in destructive vibrations shall be avoided.

- **Long-Term Shutdown and Cooling Capability**

The plant shall be shown to be capable of returning to a safe cold shutdown condition subsequent to experiencing an ATWS event, i.e., it must be shown that the reactor can be brought to a subcritical state without dependence on control rod insertion and can be cooled down and maintained in a cold shutdown condition indefinitely.

- **Fuel Integrity**

Damage to the reactor fuel rods as a consequence of an ATWS event shall not significantly distort the core, impede core cooling and prevent safe shutdown. The number of rods which would be expected to have ruptured cladding shall be determined for the purpose of evaluating radioactive releases.

- **Primary System Integrity**

The calculated reactor coolant system pressure and temperature shall be limited such that the calculated maximum primary stress anywhere in the system boundary, except steam generator tubes, is less than that permitted by the "Level C Service Limit" as defined in Section III of the ASME Nuclear Power Plant Components Code.

In addition, the deformation of reactor coolant pressure boundary components shall be limited such that the reactor can be safely brought to cold shutdown without violating any other ATWS acceptance criterion. the integrity of steam generator tubes may be evaluated based on a conservative assessment of tests and the likely condition of the tubes over their design life.

- **Mitigating Systems Design**

Mitigating systems are those systems, including any systems, equipment, or components, normally used for other functions, relied upon to limit the consequences of anticipated transients postulated to occur without scram. These systems shall be automatically initiated when the conditions monitored reach predetermined levels and continue to perform their function without operator action unless it can be demonstrated that an operator would reasonably be expected to take correct and timely action. These systems shall have high availability and in combination with the reactor protection system shall provide two independent, separate and diverse reactivity shutdown functions. The mitigating systems shall be independent, separate and diverse from the reactor trip and control rod systems, including the drive mechanisms and the neutron absorber sections. The mitigating systems shall be designed, qualified, monitored and periodically tested to assure continuing functional capability under the conditions accompanying ATWS events including natural phenomena such as earthquakes, storms including tornadoes and hurricanes, and floods expected to occur during the design life of the plant.

4.2.2.11 1979

The TMI accident forced deferral of all NRC work on ATWS and most industry work was halted or delayed as well.

4.2.2.12 1981

Proposed rules filed in federal register Vol. 46,, No. 226:

Rule #1
Early Operating Reactors

a. Modify the control rod drive scram discharge volume.

b. Provide actuation circuitry that is separate from the reactor protection system (i.e., recirculation pump trip)

Operating Plants With Construction Permits Issued Prior to 1/1/78

Provide automatic initiation of the Standby Liquid Control system and increase its flow capacity.

New Plants and Plants With Construction Permits Issued on or After 1/1/78

Addition of high capacity neutron poison injection systems.

Rule # 2
Proposed Hendrie Rule

The essence of the Hendrie rule is that power reactor licensees would be required to implement a reliable assurance program to seek out and rectify reliability deficiencies in those functions and systems that prevent or mitigate ATWS accidents.

4.2.3 Bases for ATWS Rules

In large, modern boiling water reactors, a transient with failure to scram from full power is very likely to cause or may follow the isolation of the reactor (i.e., turbine trip or main steam isolation valve closure). If the recirculation pumps continue to run, the power level will remain high and a severe pressure excursion will take place. Even if the reactor coolant system survives the pressure surge, the very high steam flow will rapidly heat the suppression pool and pressurize the containment. In addition, the High Pressure coolant Injection (HPCI) System may not suffice to cool the core: overheating and core damage may follow. Ultimately the containment is expected to rupture due to over pressure while the core sustains damage. Continued core coolant replenishment is questionable after containment rupture. A large radiological release is a plausible outcome. A necessary mitigating feature is thus a prompt automatic trip of the recirculation pumps to avoid the pressure excursion and diminish the power and the consequent steam flow to the suppression pool.

Given a trip of the recirculation pumps, the reactor power will stabilize at roughly 30% power until the reactor coolant boils down and steam bubbles (void formation) in the core throttle the chain reaction. Thereafter, a oscillatory equilibrium will be maintained in which the reactor sustains the average power necessary to boil off however much reactor coolant is delivered up to about 30% power. Analysis shows that HPCI or main feedwater can adequately cool the core to avoid extensive core damage. However, the power delivered to the suppression pool will be greater than the pool cooling system can dissipate. Therefore, containment over pressure failure remains a distinct possibility unless the reactor is shutdown, either by control rod insertion or by liquid reactivity poison injection. Well before the containment is significantly pressurized, the suppression pool will approach saturation and steam condensing will become unstable. Chugging steam condensing may threaten containment integrity or pressure suppression and thus shorten the time available to shutdown the reactor without unacceptable consequences. The HPCI is a single-train system.

The fault or human error that precipitates the initial transient might also disable the HPCI. In addition, system reliability analyses have indicated that HPCI may fail or be unavailable in as many as from 1% to 10% of the cases in which a demand is made of the system. This may be insufficient reliability for the mitigation of a potentially serious accident having a frequency of occurrence that might be as high as once in a thousand reactor years. A second diverse system, the Reactor Core Isolation Cooling (RCIC) System should be expected to auto start and run, delivering coolant to the reactor. If RCIC is the sole operative means of replenishing reactor coolant, the adequacy of core cooling, rather than the heat deposited in the suppression pool, is likely to be the factor limiting the time allowed to shut down the reactor without unacceptable consequences. The RCIC can successfully cool the reactor once it is shut down, and it can slow the boil off of reactor coolant in the reactor.

The NRC has concluded that the liquid reactivity poison injection system in large modern BWRs must have a start time and poison injection rate such that either of two redundant trains of high pressure reactor coolant replenishment systems, either of which may be expected to be available under ATWS conditions, can successfully mitigate ATWS transients. The two trains may be the HPCI and RCIC.

Concern has been expressed that the RCIC, though capable of meeting these success criteria, does not prevent the automatic depressurization of the reactor coolant system. Operator action is necessary in less than ten minutes to override the automatic depressurization. The NRC staff does not wish to force an alteration of the logic governing the Automatic Depressurization system (ADS) which might compromise the reliability of the ADS in non-ATWS events.

Several factors complicate the analysis of the ATWS tolerance of BWR plants. The delivery of main feedwater which may be available in some ATWS accident sequences may dilute liquid poison and increase the power level in ATWS events, thus threatening successful mitigation. In some sequence variants, operators might be tempted to depressurize the reactor to enable low pressure reactor coolant injection but, in so doing, disable turbine-driven coolant injection systems or otherwise compromise possible avenues of successful ATWS mitigation.

4.2.4 10CFR 50.62 (3), (4)

The Code of Federal Regulations requires all BWRs to have an alternate rod injection (ARI) system that is diverse (from the reactor trip system) from sensor output to the final actuation device. The ARI must have redundant scram air header exhaust valves. The ARI must be designed to perform its function in a reliable manner and be independent (from the existing reactor trip system) from sensor output to the final actuation device.

Each BWR must have a standby liquid control system (SLC) with the capability of injection into the reactor vessel of a borated water solution at such a flow rate, level of boron concentration and boron-10 isotope enrichment, and accounting for reactor pressure vessel volume, that the resulting reactivity control is at least equivalent to that resulting from the injection of 86 gallons per minute of 13 weight percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside diameter reactor vessel for a given core design. The SLC system and its injection location must be designed to perform its function in a reliable manner. The SLC initiation must be automatic for plants granted a construction permit prior to July 26, 1984, and for plants granted a construction permit prior to July 26, 1984, that have already been designed and built to include this feature.

Each BWR must have equipment to trip the recirculation pumps automatically under conditions indicative of an ATWS.

4.2.5 PRA Insight

The NRC staff evaluation of ATWS in NUREG-0460 was one of the first applications of PRA techniques to an Unresolved Safety Issue (USI). The evaluation highlighted the relative frequency of severe ATWS events for various reactor types and estimated the expected reduction in frequency for various postulated plant modifications. The study also proposed quantitative goals for resolving this issue. Other notable examples of PRA applications to the ATWS issue are the NRC sponsored survey and critique of reactor protection system (SAI,1982), and the ATWS Task Force report summarized in SECY-83-293. The RPS survey reviewed 16 reliability studies, most of them published PRAs, to compare the predicted failure probability per unit demand, the anticipated transient frequency, and the primary influences on RPS unavailability. There was a surprising degree of agreement among the 16 studies. The second study quantified the relative improvement to be gained by implementing a set of recommendations proposed by the utility consortium in an ATWS petition to the NRC. The third study, a value impact evaluation of the risk reduction of generic plant classes, provided the basis for a final rule on ATWS (SECY-83-293).

NUREG-1150 looked at several accident sequences which include a failure of the reactor protection system. One of the major sequences is initiated by a transient that requires a reactor scram. The mechanical RPS fails which eliminates any possibility of scramming the reactor or manually inserting control rods. The recirculation pumps are tripped and the SRVs properly cycle to control reactor pressure. The standby liquid control system is initiated manually to inject borated water into the reactor to reduce reactivity. The ADS valves are not inhibited and the reactor depressurizes which allows low pressure cooling systems to operate. The RHR system is placed in the suppression pool cooling mode or containment spray mode for containment overpressure protection, resulting in a safe core and containment.

An ATWS does have the possibility of leading to a core damage situation if the operator does not follow the Emergency Operating Procedures and initiate corrective actions like SLC initiation. However, the total contribution to the core damage frequency may not be very large (31% at Peach Bottom to 6% at Fermi).

Table 4.2-1 MSIV Closure With no Operator Action

EVENT	TIME (MIN)	COMMENT
MSIV closure initiated	0	No scram
ATWS-RPT	0.1	At reactor 1135 psia pressure
HPCI and RCIC Start	1.0	At reactor vessel level of 476.5 inches.
HPCI suction shift	8.3	High suppression pool level
HPCI fails	14.8	Suppression pool temperature 190 °F
CS and RHR systems start, ADS timer initiated.	16.0	Reactor vessel level 413.5 inches
TAF uncovered	16.7	Vessel level 360 inches
ADS actuation	18.0	Two minutes after timer actuation
BAF uncovered	19.0	Vessel level 216 inches
RHR, CS, and Condensate booster pumps start injecting	19.6	CBPs at 418 psia; CS at 357 psia; RHR at 346 psia
First core recovery	19.9	
Water introduction by RHR, CS and CBPs stop as vessel pressure increases	20.4	
Vessel pressure at relief valve setpoint	20.7	Vessel pressure 1120 psia
First core power peak	20.7	Thermal power = 178%
Drywell coolers fail on over temperature	22.4	Drywell temperature 200 °F
Second core uncovery	23.1	

Table 4.2-1 MSIV Closure With no Operator Action

EVENT	TIME (MIN)	COMMENT
RHR, CS and CBPs start injecting	24.4	
Second core recovery	24.7	
Injection by RHR, CS and CBPs stops	25.2	
Vessel pressure at relief valve setpoint	25.4	
RCIC turbine trip on high exhaust pressure	26.0	
Second power peak	27.7	Thermal power = 140 %
Third core uncovery	28.6	
RHR, CS and CBPs start injecting	29.0	
Third core uncovery	29.4	
Injection by RHR, CS and CBPs stop	29.8	
Third power peak	30.0	Thermal power = 156 %
Relief valves lift	30.1	
Fourth core uncovery	32.1	
RHR, CS and CBPs start injecting	33.6	
DRYWELL FAILS	36.8	Pressure at 132 psia

Table 4.2-2 MSIV Closure With Operator Action and With Failures of SLC and Manual Rod Insertion

EVENT	TIME (MIN)	COMMENT
MSIV closure initated	0	No scram
SRVs cycling	0 - End	No manual SRV actaution
ATWS-RPT	0.1	Reactor Vessel Pressure 1135 psia
HPCI and RCIC automatically start	1.0	Vessel water level 476.5 inches
RCIC runs at full capacity	1-End	600 gpm
Suppression pool temperature exceeds 110°F	1.5	EPG criterion for operator initiation of SLC
Operator attempts to manually insert rods	3.0	No rod motion
Operator attempts to start SLC	5.0	Pumps inoperative
Operator trips HPCI	5.0	To reduce core power and prevent HPCI failure
CS and RHR pumps start	6.2	Vessel level 413.5 inches
Level below TAF	6.8	Emergency system range, normal level indiation off scale low
Vessel level 2/3 core height	9.5	
Vessel water level stable at 2/3 core height	9.5 - End	Upper 1/3 of core steam cooled

Table 4.2-2 MSIV Closure With Operator Action and With Failures of SLC and Manual Rod Insertion

EVENT	TIME (MIN)	COMMENT
Operators initiate both loops of suppression pool cooling.	10	Containment spray select and 2/3 core coverage override hand switches actuated
Suppression pool heat capacity temperature limit exceeded	43	Operators do not depressurize
ADS two minute timer starts	50	Drywell pressure >2.45 psig, vessel level 413 inches, and Low pressure ECCS pumps running
ADS two minute reset by operators	52 - End	Prevents actuation of ADS
Suppression pool temperature at 168 °F	60	Temperature slowly increasing
Suppression pool approaching maximum temperature	360	206 °F bulk temperature

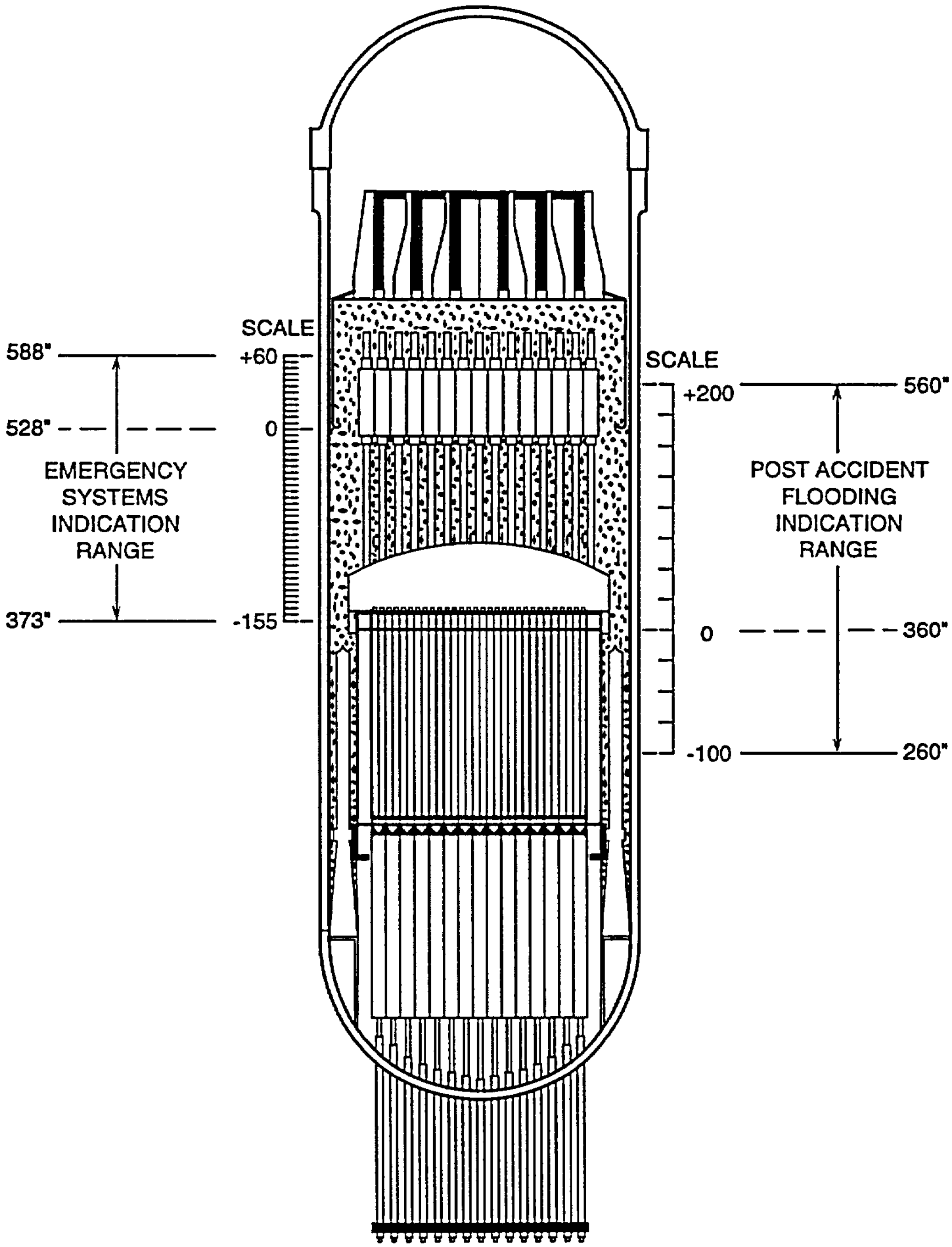

Figure 4.2-1 Level Instruments Available During ATWS Event

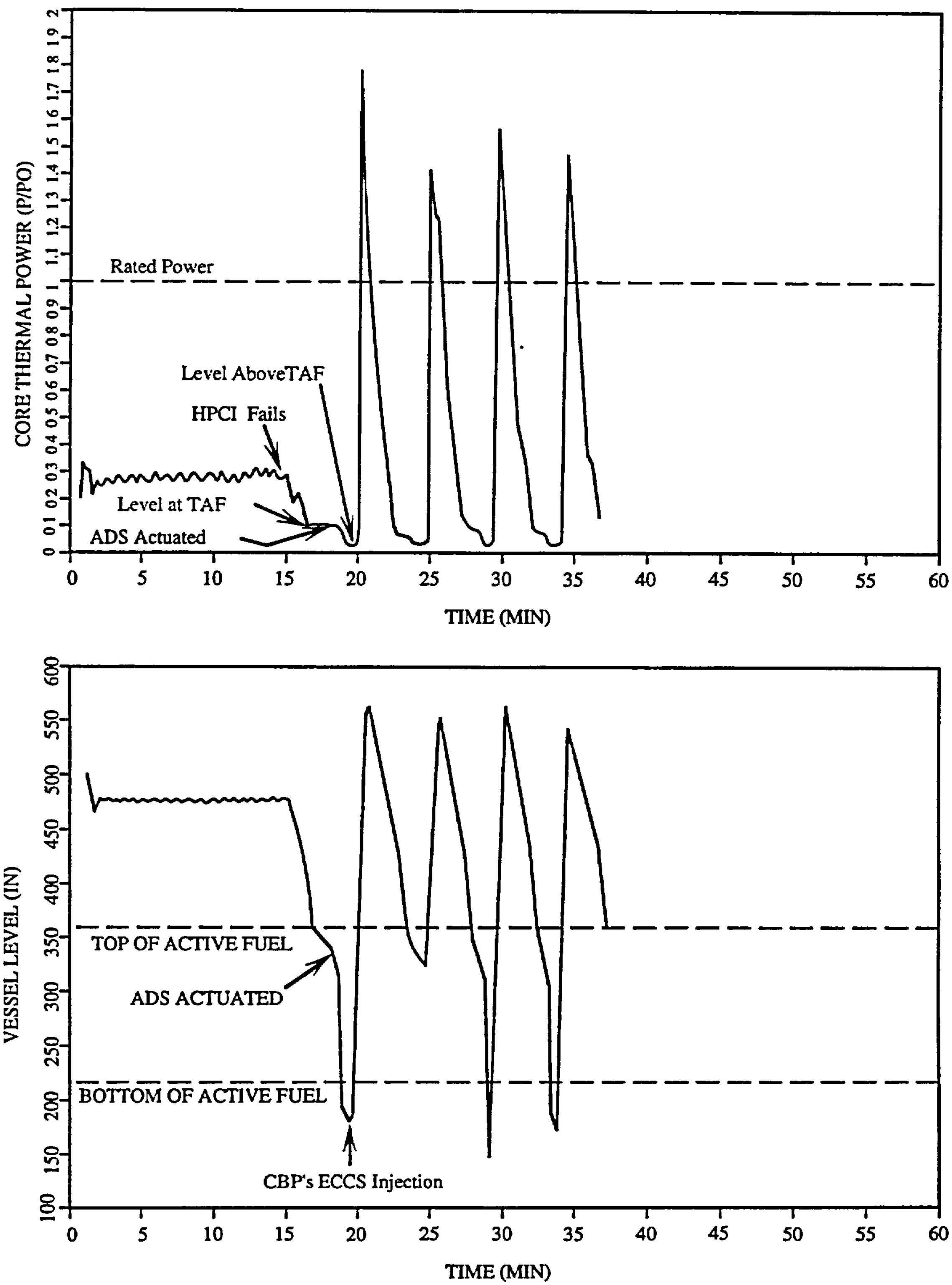

Figure 4.2-2 Reactor Vessel Level and Power

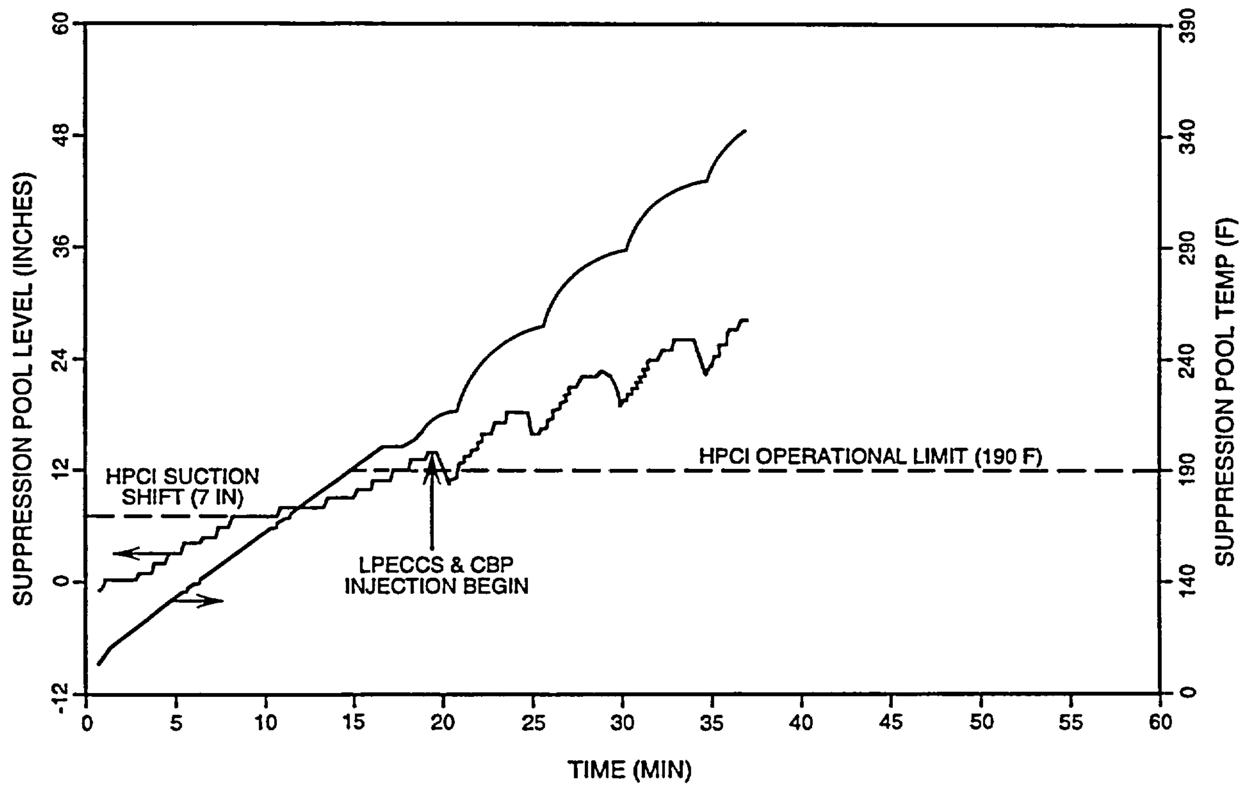

Figure 4.2-3 Suppression Pool Temperature and Level

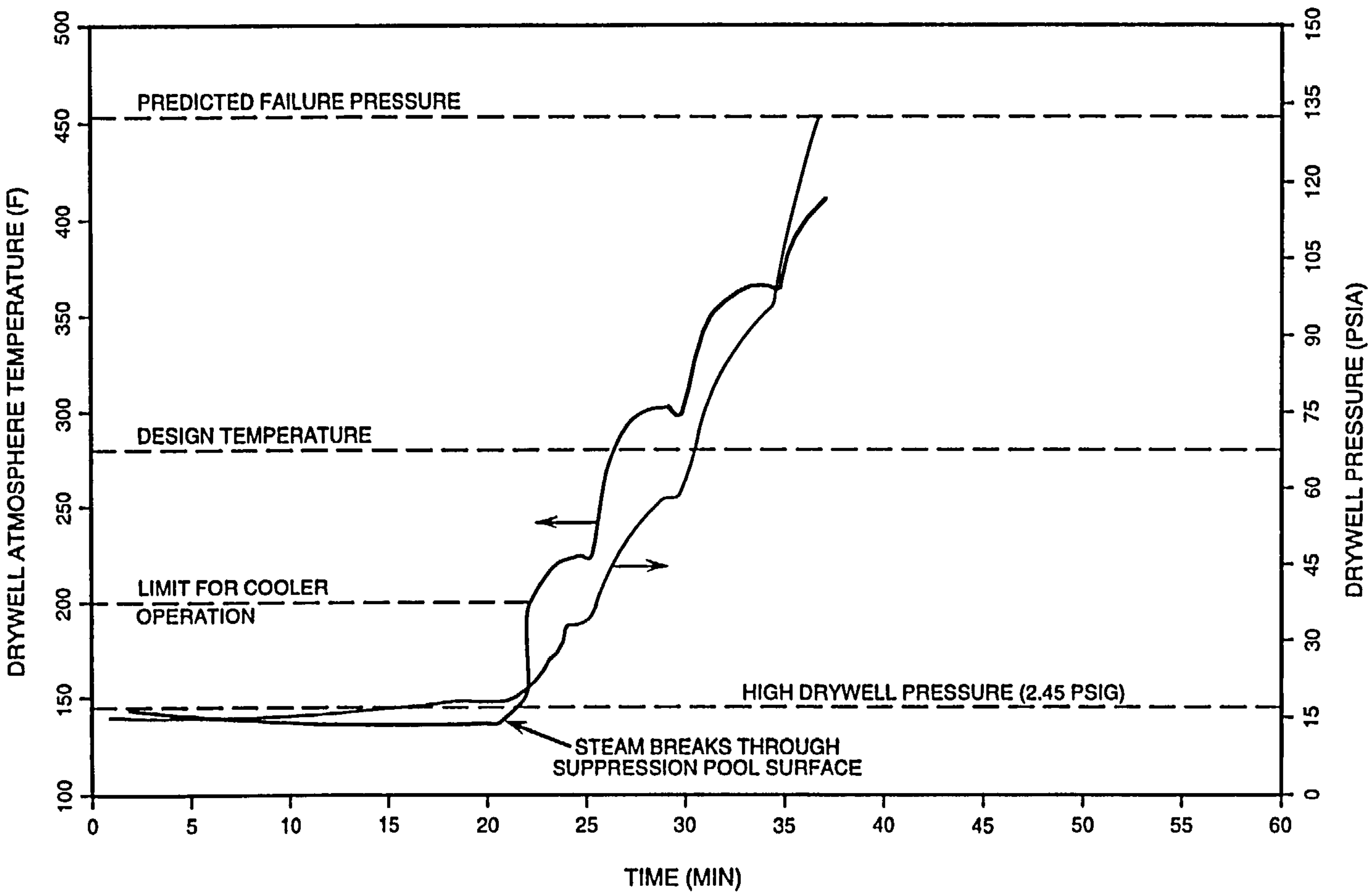

Figure 4.2-4 Drywell Pressure and Temperature

0497-4